APPENDIX D
EVALUATION OF HUMAN HEALTH EFFECTS FROM FACILITY ACCIDENTS

This appendix presents the method and assumptions used for estimating potential impacts and risks to individuals and the general public from exposure to releases of radioactive and hazardous chemical materials during hypothetical accidents at the proposed reactor facilities. The impacts from accidental radioactive material releases are given in Section D.1, and the impacts from releases of hazardous chemicals are provided in Section D.2.

D.1 RADIOLOGICAL ACCIDENT IMPACTS ON HUMAN HEALTH

D.1.1 Accident Scenario Selection and Description

D.1.1.1 Accident Scenario Selection

This accident analysis assessment considers a spectrum of potential accident scenarios. The range of accidents considered includes reactor design-basis accidents, nonreactor design-basis accidents, tritium-producing burnable absorber rod (TPBAR) handling accidents, transportation cask handling accidents, and beyond design-basis accidents (i.e., severe reactor accidents).

The spectrum of reactor and nonreactor design-basis accidents presented in the Watts Bar, Sequoyah, and Bellefonte Safety Analysis Reports were reviewed for evaluation in this environmental impact statement (EIS). The large break loss-of-coolant accident was selected as the representative reactor design-basis accident because it has the potential to damage more TPBARs than any other reactor design-basis accident (see Section D.1.1.2). Based on assumptions used in this EIS for the postulated accident scenario, the waste gas decay tank failure accident was selected as the nonreactor design-basis accident for evaluation in this EIS because it has the potential to release more tritium than other nonreactor design-basis accidents.

Following irradiation in the reactor's tritium production core, the fuel assemblies and the TPBAR assemblies inserted into the fuel assemblies would be removed from the reactor and transferred to the spent fuel pool. There, the TPBAR assemblies would be removed from the fuel assemblies. Next, the TPBARs would be removed from the TPBAR assemblies and inserted in a consolidation container. The consolidation container is a 17 × 17 array of tubes that holds the TPBARs. The consolidation container has the same footprint as a fuel assembly and can accommodate up to 289 TPBARs.

Three TPBAR handling accident scenarios are evaluated. Scenario 1 postulates that the consolidation container with 289 TPBARs is dropped while loading into a transportation cask. The evaluation further postulates that, if the consolidation container lands vertically on the spent fuel pool floor, no TPBARs would be damaged by the impact. If, however, the consolidation container lands on an edge or strikes an object (e.g., an unoccupied fuel rack or the shelf in the cask loading pit), the consolidation container shell and up to one row of tubes containing TPBARs could be damaged, and up to 17 TPBARs possibly could be breached.

Scenario 2 postulates that an irradiated fuel assembly with a TPBAR assembly containing 24 TPBARs is dropped in the spent fuel pool. The evaluation also postulates that, if the fuel assembly lands vertically, no TPBARs would be damaged by the impact. If the assembly lands on an edge or is struck by an object on the side or corner of the fuel assembly, up to 3 TPBARs could be damaged by the impact.

Scenario 3 postulates that a TPBAR assembly containing 24 TPBARs is dropped in the spent fuel pool as it is being removed from an irradiated fuel assembly and all TPBARs are breached by the impact. Scenario 3 was selected for evaluation in this EIS because it has the potential to damage more TPBARs than the other postulated TPBAR handling accidents.

Two truck or rail transportation cask drop accidents that could cause a release of tritium from the casks are evaluated in this EIS. The evaluations consider: (1) cask drops before the cask is sealed, and (2) drops that could breach a sealed cask.

The postulated beyond design-basis reactor accident analyses selected for use in this EIS address core damage accident scenarios leading to the loss of containment integrity. This includes scenarios that fall into three performance categories: (1) early containment failures, (2) late containment failures, and (3) containment bypass. Accident scenarios that do not fall into these categories lead to significantly lower consequences and, therefore, are not evaluated.

D.1.1.2 Reactor Design-Basis Accident

A reactor design-basis accident is designated as a Condition IV occurrence. Condition IV occurrences are faults that are not expected to take place, but are postulated because they have the potential to release significant amounts of radioactive material. The postulated reactor design-basis accident for this EIS is a large break loss-of-coolant accident. This postulated accident has the potential to damage more TPBARs than any other reactor loss-of-coolant design-basis accident (WEC 1998a). This accident scenario postulates a double-ended rupture of a pipe greater than 15 centimeters (6 inches) in diameter in the reactor coolant system. During the initial phase of the accident, the reactor water (coolant) level would drop below the top of the reactor core for a short period of time before the emergency systems would automatically inject additional water to cover the core. During this period the core would overheat, and the cladding on some of the fuel rods and 100 percent of the TPBARs would be breached due to the overheating (WEC 1998b). The analysis assumes that the entire tritium content in the TPBARs would be released to the containment. Each TPBAR produces 1 gram of tritium on average through the 18-month irradiation cycle (DOE 1996). For the purpose of analyses in this EIS, 1 gram of tritium contains 9,640 Curies (CRC 1982). The analysis also assumes that all of the tritium released to the reactor coolant system from the TPBARs during 17 months of normal operation would be released to the containment during the accident. This would include the release of an amount of tritium corresponding to 1 Curie per TPBAR per year (PNNL 1997). The accident consequence calculations consider applicable, reactor site-specific, protective action guidelines.

Table D–1 shows the total source term released to the containment that would be attributable to 1,000 TPBARs and a maximum of 3,400 TPBARs in a tritium production core configuration. **Table D–2** presents the tritium source term released from the containment to the environment. The reduction in the amount of tritium available for release would be the result of post-accident processing of the containment atmosphere to reduce iodine leakage to the environment, operation of hydrogen recombiners, and absorption of elemental and oxidized tritium by water in the containment (WHC 1991). In the design-basis accident, tritium would be released from the containment to the atmosphere through containment leakage. Release pathways from the containment are discussed in Section D.1.2.5.2. The analysis assumes tritiated water vapor would be released to the atmosphere for 30 days following the accident. After 30 days, all the tritiated water vapor in the containment atmosphere would be condensed and, therefore, would not be available for further release. **Table D–3** presents the accident frequency estimates.

Table D–1 Reactor Design-Basis Accident Tritium Inventory

Source Term	Tritium Production: 1,000 TPBARs (Curies)	Tritium Production: Maximum - 3,400 TPBARs (Curies)
TPBARs breached during accident	9.64×10^6	3.28×10^7
TPBAR leakage during normal operations	1,500	5,100
Total released to containment	9.64×10^6	3.28×10^7
Total available to be released to environment [a]	964,000	3.28×10^6

[a] All tritium released to the environment is in oxide form.

Table D–2 Reactor Design-Basis Accident Tritium Source Term Released to Environment

Accident Site	Tritium Production	Tritium Released (Curies) [a, b]		
		0-24 Hours	24-720 Hours	Total 0-30 Days
Watts Bar	1,000 TPBARs	814	10,700	11,600
	3,400 TPBARs	2,780	36,600	39,400
Sequoyah	1,000 TPBARs	890	11,900	12,800
	3,400 TPBARs	3,040	40,500	43,500
Bellefonte	1,000 TPBARs	338	3,880	4,220
	3,400 TPBARs	1,150	13,200	14,400

[a] All tritium released to the environment is in oxide form.
[b] Source terms for a single reactor.

Table D–3 Reactor Design-Basis Accident Frequency Estimates for Large Break Loss-of-Coolant Accident

Reactor Site	Frequency (per year)
Watts Bar	0.0002[a]
Sequoyah	0.0002[b]
Bellefonte	0.0002[c]

[a] TVA 1992b.
[b] TVA 1992a.
[c] Value currently assigned in Individual Plant Examinations.

D.1.1.3 Nonreactor Design-Basis Accident

The waste gas decay tank rupture, a Condition III occurrence, was selected as the nonreactor design-basis accident for this EIS. The consequences of a Condition III occurrence would be less severe than for a Condition IV occurrence. The release of radioactivity would not be sufficient to interrupt or restrict public use of those areas beyond the exclusion area radius (TVA 1996). The frequency of design-basis accidents is normally expected to be in the range of 0.0001 to 0.01 per year. For the purpose of this EIS, the accident frequency is assumed to be 0.01, the high end of the range.

The gaseous waste processing system is designed to remove fission product gases from the reactor coolant. The maximum storage of waste gases occurs before a refueling shutdown, at which time the gas decay tanks store the radioactive gases that are stripped from the reactor coolant. The accident analysis conservatively assumes that 10 percent of the TPBAR-generated tritium in the reactor coolant, as well as radioactive xenon and krypton fission product gases, would be stripped from the reactor coolant before a refueling shutdown and stored in waste decay tanks. Therefore, it has the potential to release more tritium than other nonreactor design-basis accidents. This assumption is conservative because the analysis postulates that all of the tritium released from the TPBARs to the reactor coolant during the entire fuel cycle would be retained in the coolant.

The postulated nonreactor design-basis accident is defined as an unexpected, uncontrolled release of the gases contained in a single gas decay tank due to the failure of the tank or the associated piping. The analysis assumes that tritium would be released directly to the environment in an oxide form. Accident consequence calculations consider applicable reactor site-specific protective action guidelines. **Table D–4** presents the tritium source term that would be released to the environment.

Table D–4 Nonreactor Design-Basis Accident Tritium Source Term

Source Term (Curies of tritium)	
1,000 TPBARs	***3,400 TPBARs***
150	510

D.1.1.4 TPBAR Handling Accident

The TPBAR handling accident scenario postulates that a TPBAR assembly containing 24 TPBARs was dropped when removing the assembly from an irradiated fuel assembly during the TPBAR consolidation process. The evaluation postulates that all TPBARs would be unprotected and would breach when they impact the spent fuel pool floor. The gaseous tritium in the 24 breached TPBARs would be released into the fuel pool and directly to the environment. The analysis conservatively assumes that the entire tritium inventory in the 24 breached TPBARs (231,360 Curies) would be released into the fuel pool (PNNL 1999). The released tritium would be in oxide form. It also was assumed that all the tritium released to the fuel pool would be released to the environment continuously over a one-year period by evaporation from the fuel pool and would be exhausted by the area ventilation system through the auxiliary building stack. This assumption was made to estimate the maximum dose to the public from this accident. [Release of tritium through liquid effluents would result in a public dose, which is an order of magnitude lower than that from release to the air.] Should a TPBAR handling accident occur, action will be taken to limit the tritium release from the breached TPBARs. However, the analysis took no credit for mitigating actions to limit the release of tritium to the fuel pool (i.e., placing the breached TPBARs in a sealed container) or to reduce the accident consequences to the public (i.e., interdiction of contaminated food and/or drinking water). **Table D–5** presents the accident frequency estimates. The frequency estimates are derived from data presented in NUREG/CR-4982, *Severe Accidents in Spent Fuel Pool in Support of Generic Safety Issue 82* (NRC 1987).

Table D–5 TPBAR Handling Accident Frequency Estimates

Frequency (per year)	
1,000 TPBARs	***3,400 TPBARs***
0.0017	0.0058

D.1.1.5 Truck Transportation Cask Handling Accident at the Reactor Site

The truck cask would be loaded under water in the spent fuel pool cask loading pit. A single TPBAR consolidation container containing a maximum of 289 TPBARs would be loaded into the cask. For the purpose of this EIS, the analysis postulates that, following insertion of the consolidation container, the cask cover would be installed but not tightly sealed. The cask would be raised above the water level where it would be hosed down and drained before moving it to the decontamination area. There it would be sealed, backfilled with inert gas, and decontaminated before loading on the truck trailer bed.

The evaluation also considered an option to seal the cask cover before lifting the cask; in this case the only potential for a tritium release would be if the cask were breached by the drop. The truck cask is designed in accordance with the requirements of 10 CFR 71, and is required to withstand a 9.1-meter (30-foot) drop onto an unyielding surface without loss or dispersal of the radioactive contents of the cask. The cask could drop more than 9.1 meters (30 feet) in the spent fuel pool cask loading pit. It could fall approximately 2.7 meters (9 feet) through the air and approximately 12.2 meters (40 feet) through the water. The terminal velocity of such a fall would exceed that reached in a 9.1 meter (30 foot) drop through air (TVA 1996). The analysis assumes that the cask would be breached by such a fall.

Spent fuel pool designs were reviewed to determine if there were any potential for cascading effects of the cask drop that would initiate releases of additional radionuclides. In the event that the spent fuel pool liner in the cask pit area is breached and the water level in the spent fuel pool drops, the water level would not drop to a level that would uncover the spent fuel in the storage racks. The cask loading area of the spent fuel pool is separated from the storage area by a shelf. The shelf height maintains the water level in the spent fuel pool storage area above the top of the spent fuel when the cask pit area is drained. Additional defense-in-depth is provided when the spent fuel pool gates are installed after loading the cask. With the gates in place, one on each side of the cask loading pit access channel to the spent fuel pool, a breach of the liner in the cask loading pit area would result in a drop in the spent fuel water level to the top of the gates.

The analysis assumed that, in the event the cask is dropped onto the floor of the fuel pool area, the cask would not penetrate the floor or damage equipment located at an elevation below the potential drop zone. Analyses would be performed, if necessary, to verify this assumption during the U.S. Nuclear Regulatory Commission (NRC) operating license process and/or license amendment process.

It is anticipated that no TPBARs would be damaged by the drop. The TPBARs in the cask would be protected from damage not only by the cask, but also by the consolidation container structure. However, the analysis conservatively assumes that the structural loads on the TPBARs resulting from the drop could breach up to 17 TPBARs, the same number considered for a dropped TPBAR consolidation container. The gaseous tritium in the 17 breached TPBARs would be released into the fuel pool and directly to the environment by evaporation. Two accident scenarios are considered. Scenario 1 assumes that the cask drop occurs prior to draining and drying the cask interior. The analysis conservatively assumes that the 17 breached TPBARs release tritium into the flooded cask at the rate of 50 Curies per TPBAR per day (PNNL 1999) until the cask can be drained into the fuel pool and the cask interior can be vacuum-dried. The analysis further assumes that the cask is drained and vacuum-dried within seven days of the accident to limit the release of tritium from the breached TPBARs. The analysis takes no credit for additional mitigating actions to reduce the released tritium to the fuel pool (e.g., draining the cask into a storage tank). A total of 5,950 Curies of tritium, in oxide form, would be released to the fuel pool area and exhausted up the auxiliary building stack over a one-year period.

Scenario 2 assumes that the cask drop of more than 30 feet occurs while loading the cask onto a trailer after it is loaded with TPBARs, sealed, and decontaminated. It is assumed that this accident would result in 17 breached TPBARs and loss of the cask confinement integrity. The breached TPBARs would release tritium, assumed to be in oxide form, to the auxiliary building atmosphere at a rate of 0.00001 grams per breached TPBAR per hour (PNNL 1999). Further, the analysis assumes that the tritium release would be terminated when the TPBARs are placed in a replacement cask within 30 days of the accident. During this period, a total of 1,180 Curies of tritium would be released to the atmosphere through the auxiliary building stack. The consequences for Scenario 1 bound the consequences of Scenario 2.

Table D–6 presents the frequency estimates for the truck transportation cask handling accident (Scenario 1). The frequency estimates are derived from data presented in NUREG/CR-4982, *Severe Accidents in Spent Fuel Pool in Support of Generic Safety Issue 82* (NRC 1987).

Table D–6 Truck Transportation Cask Handling Accident Frequency Estimates

Frequency (per year)	
1,000 TPBARs	***3,400 TPBARs***
5.3×10^{-7}	1.6×10^{-6}

D.1.1.6 Truck Transportation Cask Handling Accident at the Tritium Extraction Facility

Cask handling accidents at the Tritium Extraction Facility are in the scope of the Tritium Extraction Facility EIS and are not within the scope of this EIS.

D.1.1.7 Rail Transportation Cask Handling Accident at the Reactor Site

The rail cask would be loaded under water in the spent fuel pool cask loading pit with 3 to 12 TPBAR consolidation containers. For the purpose of this EIS, the analysis postulates that, following insertion of the consolidation containers, the cask cover would be installed, but not tightly sealed. The cask would be raised above the water level, where it would be hosed down, drained, and the cask interior would be vacuum-dried before moving it to the decontamination area. There it would be sealed, backfilled with inert gas, and decontaminated before loading on the rail car.

The evaluation also considers an option to seal the cask cover before lifting the cask; in this case the only potential for a tritium release would be if the cask were breached by the drop. The rail cask is designed in accordance with the requirements of 10 CFR 71, which requires that the cask withstand a 9.1-meter (30-foot) drop onto an unyielding surface without loss or dispersal of the radioactive contents of the cask. The cask could drop more than 9.1 meters (30 feet) in the spent fuel pool cask loading pit. Here the cask could fall approximately 2.7 meters (9 feet) through air and approximately 12.2 meters (40 feet) through water. The terminal velocity reached in such a fall would exceed that reached in a 9.1-meter (30-foot) drop through air (TVA 1996). The analysis assumes that the cask would be breached by such a fall.

Spent fuel pool designs were reviewed to determine if there were any potential for cascading effects of the cask drop that would initiate releases of additional radionuclides. In the event that the spent fuel pool liner in the cask pit area is breached and the water level in the spent fuel pool drops, the water level would not drop to a level that would uncover the spent fuel in the storage racks. The cask loading area of the spent fuel pool is separated from the storage area by a shelf. The shelf height maintains the water level in the spent fuel pool storage area above the top of the spent fuel when the cask pit area is drained.

The analysis assumes that, in the event the cask is dropped onto the floor of the fuel pool area, the cask would not penetrate the floor or damage equipment located at an elevation below the drop zone. Analyses will be performed to verify this assumption during the NRC operating license process and/or license amendment process.

It is anticipated that no TPBARs would be damaged by the drop. The TPBARs in the cask would be protected from damage not only by the cask, but also by the TPBAR consolidation container structure. However, the analysis conservatively assumes that the structural loads on the TPBARs resulting from the drop could breach up to 17 TPBARs, the same number considered for a dropped TPBAR consolidation container. Two accident scenarios are considered. Scenario 1 assumes that the cask drop occurs prior to draining and drying the cask interior. The analysis conservatively assumes that the 17 breached TPBARs release tritium into the flooded cask at the rate of 50 Curies per TPBAR per day (PNNL 1999) until the cask can be drained into the fuel pool and the cask interior can be vacuum-dried. The analysis further assumes that the cask is drained and dried within seven days of the accident to limit the release of tritium from the breached TPBARs. The analysis takes no credit for additional mitigating actions to reduce the released tritium to the fuel pool (e.g., draining the cask into a storage tank). A total of 5,950 Curies of tritium, in oxide form, would be released to the fuel pool area and exhausted up the auxiliary building stack over a one-year period.

Scenario 2 assumes that the cask drop of more than 30 feet would occur while loading the cask onto a rail car after it is loaded with TPBARs, sealed, and decontaminated. It is assumed that this accident would result in 17 breached TPBARs and loss of the cask confinement integrity. The breached TPBARs would release tritium, assumed to be in oxide form, to the auxiliary building atmosphere at a rate of 0.00001 grams per breached TPBAR per hour (PNNL 1999). Further, the analysis assumes that the tritium release would be terminated when the TPBARs are placed in a replacement cask within 30 days of the accident. During this period, a total of 1,180 Curies of tritium would be released to the atmosphere through the auxiliary building stack. The consequences for Scenario 1 bound the consequences of Scenario 2.

Table D–7 presents the frequency estimates for the rail transportation cask handling accident (Scenario 1). The frequency estimates are derived from data presented in NUREG/CR-4982, *Severe Accidents in Spent Fuel Pool in Support of Generic Safety Issue 82* (NRC 1987), and the assumption that each rail cask would contain three TPBAR consolidation containers.

Table D–7 Rail Transportation Cask Handling Accident Frequency Estimates

Frequency (per year)	
1,000 TPBARs	***3,400 TPBARs***
2.7×10^{-7}	8.0×10^{-7}

D.1.1.8 Rail Transportation Cask Handling Accident at the Savannah River Site Rail Transfer Station

Rail service is provided on DOE's Savannah River Site in South Carolina, but not directly to the Tritium Extraction Facility. Rail casks would be transferred to a truck at an onsite rail transfer station for transport to the Tritium Extraction Facility. The rail cask is designed in accordance with the requirements of 10 CFR 71, which requires that the cask be able to withstand a 9.1-meter (30-foot) drop onto an unyielding surface without loss or dispersal of the radioactive contents of the cask. During transfer of the cask from the rail car to the truck, the cask elevation above the ground would not exceed 9.1 meters (30 feet). Therefore, postulated cask handling accidents at the rail transfer station (i.e., cask drop events) would not cause breach of the cask and release of the radioactive material.

D.1.1.9 Rail Transportation Cask Handling Accident at the Tritium Extraction Facility

Cask handling accidents at the Tritium Extraction Facility are in the scope of the Tritium Extraction Facility EIS and are not within the scope of this EIS. The scope of the Tritium Extraction Facility EIS starts with the delivery of irradiated TPBARs at the Tritium Extraction Facility.

D.1.1.10 Beyond Design-Basis Accident

The beyond design-basis accident is limited to the severe reactor accidents. Severe reactor accidents are less likely to occur than reactor design-basis accidents. The consequences of these accidents could be more serious if no mitigative actions are taken. In the reactor design-basis accidents, the mitigating systems are assumed to be available. In the severe reactor accidents, even though the initiating event could be a design-basis event (e.g., large break loss-of-coolant accident), additional failures of mitigating systems would cause some degree of physical deterioration of the fuel in the reactor core and a possible breach of the containment structure leading to releases of radioactive materials to the environment. For the purposes of this EIS, only the severe reactor accident scenarios that lead to containment bypass or failure are considered. Accident scenarios that do not lead to containment bypass or failure are not presented because the public and environmental consequences would be significantly less in those cases. It should be noted that analyses performed as part of the New Production Reactor program in the late 1980s concluded that severe accident core melts do not lead to uncontrolled recriticality if the core enrichment is less than 7.5 percent. Since CLWR core enrichments are less than 5 percent, recriticality is not considered.

In 1988, the NRC asked all licensees of operating plants to perform individual plant examinations for severe accident vulnerabilities (NRC 1988). In the request, the NRC indicated that a probabilistic risk assessment is an acceptable approach to use in performing the individual plant examination. This analysis evaluates in full detail (quantitatively) the consequences of all potential events caused by the operating disturbances (known as internal initiating events) within each plant. [See the discussion under severe reactor accident scenarios presented below.] The state-of-the-art probabilistic risk assessment uses realistic criteria and assumptions in evaluating the accident progression and the systems required to mitigate each accident.

In 1991, the NRC requested that all licensees of operating plants should conduct individual plant examinations of external events for severe accident vulnerabilities (NRC 1991). This analysis covers the accidents that could be initiated naturally (e.g., earthquakes, tornadoes, floods, strong winds) and/or manmade (e.g., aircraft crash and fire). The individual plant examination of external event analyses are less quantitative and results-oriented than those performed under individual plant examination. The analyses were done to confirm that no vulnerabilities or issues exist and that the plants would have sufficient capacity to continue functioning in beyond design-basis external events.

Currently, plant-specific severe accident analyses are only available for operating plants such as the Sequoyah and Watts Bar Nuclear Plants. No such analyses are available for the Bellefonte Nuclear Plant. However, the results of such studies will be available prior to operation of the Bellefonte Nuclear Plant.

Severe Reactor Accident Scenarios

Before identifying the accident scenarios that lead to failure of the containment, it is important to provide a brief overview of the present severe accident analysis techniques used in plant-specific probabilistic risk assessments or individual plant examinations for severe accident vulnerabilities (NRC 1990b). The analysis starts with identification of initiating events (i.e., challenges to normal plant operation or accidents) that require successful mitigation to prevent core damage. These events are grouped into initiating event classes that have similar characteristics and require the same overall plant response.

For example, a loss of offsite power to a plant could be caused by severe weather events (high wind, tornado, hurricane, and snow and ice storms), power substation breaker faults, instability in the power transmission lines, unbalanced loading of power lines, etc. Each of these events would lead to loss of main generator power and a reactor trip, which would challenge the same safety functions. These events are grouped together and analyzed under the loss of offsite power initiating event.

Event trees are developed for each initiating event class. These event trees depict the possible sequence of events that could occur during the plant's response to each initiating event class. The trees delineate the possible combinations (sequences) of functional and/or system successes and failures that lead to either successful mitigation of the initiator or core damage. Functional and/or system success criteria are developed based on the plant response to the class of accidents. Failure modes of systems that are functionally important to preventing core damage are modeled. This modeling process is usually done with fault trees that define the combinations of equipment failures, equipment outage, and human errors that cause the failure of systems to perform the desired function.

Quantification of the event trees leads to hundreds, or even thousands, of different end states representing various accident sequences that lead to core damage. Each accident sequence and its associated end state has a unique "signature" because of the particular combination of system successes and failures events. These end states are grouped together into plant damage states, each of which collects sequences for which the progression of core damage, the release of fission products from the fuel, the status of containment and its systems, and the potential for mitigating source terms are similar. The sum of all core damage accident sequences then will represent an estimate of plant core damage frequency. The analysis of core damage frequency calculations is called a level 1 probabilistic risk assessment, or front-end analysis.

Next, an analysis of accident progression, containment loading resulting from the accident, and the structural response to the accident loading is performed. The primary objective of this analysis, which is called a level 2 probabilistic risk assessment, is to characterize the potential for, and magnitude of, a release of radioactive material from the reactor fuel to the environment, given the occurrence of an accident that damages the core. The analysis includes an assessment of containment performance in response to a series of severe accidents. Analysis of the progression of an accident (an accident sequence within a plant damage state) generates a time history of loads imposed on the containment pressure boundary. These loads then would be compared against the containment's structural performance limits. If the loads exceed the performance limits, the containment would be expected to fail; conversely, if the containment performance limits exceed the calculated loads, the containment would be expected to survive. Three modes of containment failures are defined: containment bypass, early containment failure, and late containment failure (see **Table D–8**).

The magnitude of the radioactive release to the atmosphere in an accident is dependent on the timing of the reactor vessel failure and the containment failure. To determine the magnitude of the release, a containment event tree representing the time sequence of major phenomenological events that could occur during the formation and relocation of core debris (after core melt), the availability of the containment heat removal system, and the expected mode of containment failures (i.e., bypass, early, and late), is developed. A reduced set of plant damage states are defined by culling the lower frequency plant damage states into higher frequency ones that have relatively similar severity and consequence potential. This condensed set is known as the key plant damage states (a functional sequence that either has a core damage frequency greater than or equal to 10^{-6} per reactor year or leads to containment bypass at a frequency of greater than or equal to 10^{-7} per reactor year (NRC 1988). These key plant states then would become the initiating events for the containment event tree. The outcome of each sequence in this event tree represents a specific release category. Release categories that can be represented by similar source terms are grouped. Source terms associated with various release categories describe the fractional releases for representative radionuclide groups, as well as the timing, duration, and energy of release.

Table D–8 Definition and Causes of Containment Failure Mode Classes

Failure mode	*Definition and Causes*
Containment Bypass	Involves failure of the pressure boundary between the high-pressure reactor coolant and low-pressure auxiliary system. For pressurized water reactors, steam generator tube rupture, either as an initiating event or as a result of severe accident conditions, will lead to containment bypass. In these scenarios, if core damage occurs, a direct path to the environment can exist.
Early Containment Failure	Involves structure failure of the containment before, during, or slightly after (within a few hours) reactor vessel failure. A variety of mechanisms can cause structure failure such as: direct contact of core debris with containment, rapid pressure and temperature loads, hydrogen combustion, and fuel coolant interaction (ex-vessel steam explosion). Failure to isolate containment and an early vented containment after core damage also are classified as early containment failures.
Late Containment Failure	Involves structural failure of the containment several hours after reactor vessel failure. A variety of mechanisms can cause late structure failure such as: gradual pressure and temperature increase, hydrogen combustion, and basemat melt-through by core debris. Venting containment late in the accident also is classified as a late containment failure.

Most of the current plant probabilistic risk assessment analyses end at this stage. Only a limited number of plants have performed an evaluation of resulting consequences to the public and environment from releases of radioactive materials following a core melt and containment failure. This type of analysis, which is known as a level 3 probabilistic risk assessment, was first performed by the NRC in WASH-1400 (NRC 1975). In the late 1980s, the NRC performed a comprehensive, full-scope severe accident analyses for five different plant types and documented the results in NUREG-1150 (NRC 1990b). The analyses provided in this EIS use the insights gained from this NRC report and follow the methods applied and the assumptions made to estimate the consequences to the public and the environment.

Representative Severe Reactor Accident Scenarios for the Sequoyah and Watts Bar Nuclear Plants

As stated earlier, only the plant damage states that lead to containment failure (failure mode defined as bypass, early, and late) and release of radioactive materials to the environment are considered in this EIS. The description of the representative accident scenarios is limited to the dominant sequence (or sequences) within a plant damage state that is a major contributor to the release level categories associated with each of the containment failures defined above. For Watts Bar and Sequoyah, the information is based on the most recent analysis of severe accidents performed by the Tennessee Valley Authority (TVA) under the individual plant examination program that covers both the level 1 and level 2 probabilistic risk assessments in detail. TVA's analyses of the Watts Bar and Sequoyah individual plant examinations were submitted to the NRC in September 1992 (TVA 1992a, TVA 1992b). Both of these analyses have been revised (TVA 1995b, TVA 1994), and the Watts Bar 1 analysis has been revised even further (TVA 1998).

The selected release categories and examples of various accident scenarios leading to containment failure and/or bypass are presented below for the Sequoyah and Watts Bar Nuclear Plants. **Table D–9** shows reactor core inventories for Watts Bar 1 and Sequoyah 1 and 2. **Table D–10** provides important information on time to core damage, containment failure, release duration, and the isotope release fractions associated with each of the release levels. **Table D–11** provides a representation of the dominant accident scenarios that lead to each release category, along with its likelihood of occurrence. Release Category I results from a reactor vessel breach with early containment failure. Release Category II results from a reactor vessel breach with containment bypass. Release Category III results from a reactor vessel breach with late containment failure.

Table D–9 Watts Bar 1 and Sequoyah 1 and 2 Core Inventory

Nuclide	Isotope	Inventory (Curies)
Cobalt:	Co-58	874,000
	Co-60	668,000
Krypton:	Kr-85	671,000
	Kr-85m	3.14×10^{7}
	Kr-87	5.74×10^{7}
	Kr-88	7.76×10^{7}
Rubidium:	Rb-86	51,200
Strontium:	Sr-89	9.73×10^{7}
	Sr-90	5.25×10^{6}
	Sr-91	1.25×10^{8}
	Sr-92	1.30×10^{8}
Yttrium:	Y-90	5.64×10^{6}
	Y-91	1.19×10^{8}
	Y-92	1.31×10^{8}
	Y-93	1.48×10^{8}
Zirconium:	Zr-95	1.50×10^{8}
	Zr-97	1.56×10^{8}
Niobium:	Nb-95	1.42×10^{8}
Molybdenum:	Mo-99	1.65×10^{8}
Technetium:	Tc-99m	1.43×10^{8}
Ruthenium:	Ru-103	1.23×10^{8}
	Ru-105	8.01×10^{7}
	Ru-106	2.80×10^{7}
Rhodium:	Rh-105	5.55×10^{7}
Antimony:	Sb-127	7.56×10^{6}
	Sb-129	2.68×10^{7}
Tellurium	Te-127	7.30×10^{6}
	Te-127m	966,000
	Te-129	2.51×10^{7}
	Te-129m	6.62×10^{6}
	Te-131m	1.27×10^{7}
	Te-132	1.26×10^{8}
Iodine:	I-131	8.69×10^{7}
	I-132	1.28×10^{8}
	I-133	1.84×10^{8}
	I-134	2.02×10^{8}
	I-135	1.73×10^{8}
Xenon	Xe-133	1.84×10^{8}
	Xe-135	3.45×10^{7}
Cesium:	Cs-134	1.17×10^{7}
	Cs-136	3.57×10^{6}
	Cs-137	6.55×10^{6}
Barium:	Ba-139	1.70×10^{8}
	Ba-140	1.69×10^{8}
Lanthanum:	La-140	1.72×10^{8}
	La-141	1.58×10^{8}
	La-142	1.52×10^{8}
Cerium:	Ce-141	1.53×10^{8}
	Ce-143	1.49×10^{8}
	Ce-144	9.23×10^{7}
Praseodymium:	Pr-143	1.46×10^{8}
Neodymium:	Nd-147	6.54×10^{7}
Neptunium:	Np-239	1.75×10^{9}

Nuclide	*Isotope*	*Inventory (Curies)*
Plutonium:	Pu-238 Pu-239 Pu-240 Pu-241	99,300 22,400 28,200 4.76×10^6
Americium:	Am-241	3,140
Curium:	Cm-242 Cm-244	1.20×10^6 70,400

Source: NUREG/CR-4551 (NRC 1990b)

Table D–10 Release Category Timing and Source Terms

Release Times, Heights, Energies, and Source Terms for Selected Watts Bar and Sequoyah Nuclear Plants Release Categories					
Release Category	*Release Height (meters)*	*Warning Time (hours)*	*Release Time (hours)*	*Release Duration (hours)*	*Release Energy[a] (megawatts)*
I	10.00	8	10	2	28
II	10.00	20	24	4	1
III	10.00	20	30	10	3.5

Fission Product Source Terms (fraction of total inventory)[b]										
Release Category	*NG*	*I*	*Cs*	*Te*	*Sr*	*Ru*	*La*	*Ce*	*Ba*	*Mo*
I	0.90	0.042	0.043	0.044	0.0027	0.0065	0.00048	0.004	0.0046	0.0065
II	0.91	0.21	0.19	0.0004	0.0023	0.07	0.00028	0.00055	0.025	0.07
III	0.94	0.0071	0.011	0.0052	0.00036	0.00051	$\underline{4.2} \times 10^{-6}$	4.0×10^{-6}	0.0013	0.00051

NG = Noble gases.
[a] These values were taken from similar accident scenarios as given in NUREG/CR-4551.
[b] See Table D-9 for explanations of the chemical abbreviations used for the fission products listed above.
Source: TVA 1992a, TVA 1992b.

Table D–11 Release Category Frequencies and Related Accident Sequences for the Watts Bar and Sequoyah Nuclear Plants

Watts Bar Nuclear Plant		
I	6.8×10^{-7}	The major accident contributors to this release event are initiated by loss of offsite power and loss of the essential raw cooling water system with failure of the emergency diesels to start and/or failures in the 125-volt direct current distribution system, in conjunction with loss of secondary cooling and no recovery before core melt.
II	6.9×10^{-6}	The main contributor to this release event is initiated by a steam generator tube rupture in conjunction with either an operator error or random failure of electrical distribution systems, leading to failure of the coolant system and failure to control the affected steam generator before core melt occurs.
III	9.1×10^{-6}	The major accident contributors to this release event are initiated by loss of offsite power with various failures in the alternating current distribution systems and no recovery of power before core melts, and by a reactor coolant system loss-of-coolant accident (large- and medium-sized loss-of-coolant accident) with failure to establish long-term core cooling.

Sequoyah Nuclear Plant		
Release Category	***Release Frequency***	***Representative Accident Scenario(s)***
I	6.8×10^{-7}	The major accident contributors to this release event are initiated by loss of the 125-volt battery boards and loss of all offsite power with the failure of emergency diesels to start (station blackout: loss of all alternating current power to all emergency core cooling systems), as well as the failure of the auxiliary feedwater system (loss of secondary cooling) with no recovery before core melt.
II	4.0×10^{-6}	The accident scenario for this release event is similar to that given for the Watts Bar plant, above.
III	9.2×10^{-6}	The major accident contributors to this release event are initiated by: loss of offsite power with various failures in the alternating current and/or direct current distribution systems and no recovery of power before core melt, and by reactor coolant system small break loss-of-coolant accident (caused by either loss of the component cooling system leading to development of reactor coolant pump seals failure or another nonisolatable break in the reactor coolant system) with failure to depressurize the reactor and/or establish long-term reactor core cooling.

Representative Severe Accident Scenarios for the Bellefonte Nuclear Plant

For the Bellefonte Nuclear Plant, no plant-specific severe accident analysis information is available. This plant will have a complete probabilistic risk assessment covering both the internal and the external initiating events prior to the issuance of an operating license by the NRC. For the purposes of this EIS, a surrogate list of accident scenarios will need to be selected based on the review of accident analyses of similar plants. For this selection process, the publicly available reports on individual plant examination results from Three Mile Island 1 (GPUN 1993); Arkansas Nuclear One Unit 1 (Entergy 1993); and the Oconee Nuclear Station (Duke 1990), as well as a limited scope level 1 probabilistic risk assessment (core damage frequency calculation) report on the uncompleted Washington Nuclear Plant Unit 1 (WHC 1992), were reviewed. The review process identified Washington Nuclear Plant Unit 1 as the most similar in its nuclear steam supply system and containment structure to the Bellefonte Nuclear Plant.

Based on the above review, the Washington Nuclear Plant Unit 1 limited level 1 probabilistic risk assessment report was used as a surrogate for the Bellefonte Nuclear Plant. The core damage frequency calculations in this report include the estimate for the original design as well as that for a modified safety system. For the purposes of this EIS, the core damage frequency associated with the original (as built) design was considered. For the level 2 analysis, e.g., determination of containment performance in severe accidents and corresponding release categories, the analyses presented in WHC-EP-0263 (WHC 1991) were used. Again, the release category frequencies given in this report were modified to reflect that of the original design. In addition, in order to present the release categories consistent with those given for the Watts Bar and Sequoyah Nuclear Plants, the release categories were regrouped (WHC 1991) as Release Category I, II, and III, and the bounding release fractions and the shortest timings in each group were assigned to the new release categories.

The selected release categories and examples of various accident scenarios leading to containment failure and/or bypass are presented below for the Bellefonte plant. **Table D–12** presents the reactor core inventory for the Bellefonte plant. **Table D–13** provides relevant information on time to core damage, containment failure, release duration, and the isotope release fractions associated with each of the release levels. **Table D–14** provides a brief representation of dominant accident scenarios that lead to each release category level, along with its likelihood of occurrence.

Table D–12 Bellefonte Nuclear Plant Reactor Core Inventory

Nuclide	*Isotope*	*Inventory (Curies)*
Cobalt:	Co-58	919,000
	Co-60	703,000
Krypton:	Kr-85	706,000
	Kr-85m	3.30×10^{7}
	Kr-87	6.04×10^{7}
	Kr-88	8.17×10^{7}
Rubidium:	Rb-86	53,800
Strontium:	Sr-89	1.02×0^{8}
	Sr-90	5.53×10^{6}
	Sr-91	1.32×10^{8}
	Sr-92	1.37×10^{8}
Yttrium:	Y-90	5.93×10^{6}
	Y-91	1.25×10^{8}
	Y-92	1.37×10^{8}
	Y-93	1.56×10^{8}
Zirconium:	Zr-95	1.58×0^{8}
	Zr-97	1.64×10^{8}
Niobium	Nb-95	1.49×10^{8}
Molybdenum:	Mo-99	1.74×10^{8}
Technetium:	Tc-99m	1.50×10^{8}
Ruthenium:	Ru-103	1.30×10^{8}
	Ru-105	8.42×10^{7}
	Ru-106	2.94×10^{7}
Rhodium:	Rh-105	5.83×10^{7}
Antimony:	Sb-127	7.95×10^{6}
	Sb-129	2.81×10^{7}
Tellurium	Te-127	7.68×10^{6}
	Te-127m	1.02×10^{6}
	Te-129	2.64×10^{7}
	Te-129m	6.97×10^{6}
	Te-131m	1.33×10^{7}
	Te-132	1.33×10^{8}
Iodine:	I-131	9.14×10^{7}
	I-132	1.35×10^{8}
	I-133	1.93×10^{8}
	I-134	2.12×10^{8}
	I-135	1.82×10^{8}
Xenon:	Xe-133	1.93×10^{8}
	Xe-135	3.63×10^{7}
Cesium:	Cs-134	1.23×10^{7}
	Cs-136	3.75×10^{6}
	Cs-137	6.89×10^{6}
Barium:	Ba-139	1.79×10^{8}
	Ba-140	1.77×10^{8}
Lanthanum:	La-140	1.81×10^{8}
	La-141	1.66×10^{8}
	La-142	1.60×10^{8}
Cerium:	Ce-141	1.61×10^{8}
	Ce-143	1.57×10^{8}
	Ce-144	9.71×10^{7}
Praseodymium:	Pr-143	1.54×10^{8}
Neodymium:	Nd-147	6.88×10^{7}
Neptunium:	Np-239	1.84×10^{9}

Nuclide	*Isotope*	*Inventory (Curies)*
Plutonium:	Pu-238	104,000
	Pu-239	23,600
	Pu-240	29,700
	Pu-241	5.00×10^6
Americium:	Am-241	3,300
Curium:	Cm-242	1.26×10^6
	Cm-244	74,000

Source: Derived from NUREG/CR-4551 (NRC 1990b) by multiplying the values given in Table D–9 by the 1.055 (core thermal ratio of Bellefonte over Sequoyah Nuclear Plants).

Table D–13 Release Category Timing and Source Term

Release Times, Heights, Energies, and Source Terms for Selected Bellefonte Nuclear Plant Release Categories					
Release Category	*Release Height (meters)*	*Warning Time (hours)*	*Release Time (hours)*	*Release Duration (hours)*	*Release Energy (megawatts)*
I	15	2.0	3.0	5	40
II	30	2.0	3.0	1	30
III	15	10	24	5	40

Fission Product Source Terms (fraction of total inventory)[a]										
Release Category	*NG*	*I*	*Cs*	*Te*	*Sr*	*Ru*	*La*	*Ce*	*Ba*	*Mo*
I	1.0	0.003	0.003	0.006	0.0004	3.0×10^{-6}	3.0×10^{-6}	3.0×10^{-5}	0.0002	0.0002
II	1.0	0.07	0.07	0.1	0.01	6.0×10^{-5}	6.0×10^{-5}	0.0007	0.005	0.004
III	0.7	0.001	0.001	0.007	8.0×10^{-5}	8.0×10^{-7}	8.0×10^{-7}	9.0×10^{-6}	0.0001	3.0×10^{-6}

NG = noble gases.
[a] See Table D-12 for explanations of the chemical abbreviations used for the fission products listed above.
Source: WHC 1991.

Table D–14 Release Category Frequencies and the Related Accident Sequences for the Bellefonte Nuclear Plant

Release Category	*Release Frequency*	*Representative Accident Scenario(s)*
I	9.0×10^{-7}	The major accident contributors to this release event would be initiated by a loss of offsite power with failure of the diesel generators (station blackout) and long-term failure of the auxiliary feedwater system. Containment fails early.
II	9.1×10^{-7}	The major accident contributors to this release event would be initiated by a small loss-of-coolant accident followed by failure of emergency recirculation, containment spray recirculation, and containment isolation, and by a loss of offsite power with failure of the diesel generators (station blackout) and no recovery of power before core melt and containment isolation fails.
III	5.1×10^{-6}	The major accident contributors to this release event are initiated by a loss of offsite power with failure of the diesel generators (station blackout) and long-term failure of the auxiliary feedwater system. Containment fails late.

The information presented in the preceding three tables represents the best available estimate for the core damage frequency and characteristics without a plant-specific probabilistic assessment such as those performed for the Watts Bar and Sequoyah Nuclear Plants. The Washington Nuclear Plant was selected as exhibiting the most representative design, but differences between this plant and the Bellefonte Nuclear Plant are to be expected. The referenced probabilistic analysis is a limited scope analysis and the Washington Nuclear Plant, like the Bellefonte Nuclear Plant, is not in commercial operation. [The lack of operational data results in the use of some more conservative assumptions that impact the analysis results.] However, use of this data with Bellefonte Nuclear Plant site-specific population and weather data does allow a representative calculation of risk to be performed.

D.1.2 Methodology for Estimating Radiological Impacts

D.1.2.1 Introduction

The GENII and MACCS2 computer codes were used to perform probabilistic analyses of radiological impacts. The GENII computer code was used to estimate the consequences of the reactor design-basis, nonreactor design-basis, TPBAR-handling, and cask-handling accidents. The MACCS2 computer code was used for the beyond design-basis accidents. In addition, deterministic analyses, using the method in the reactor facility safety analysis reports, were performed for the release of tritium in the reactor and the nonreactor design-basis accidents. This additional analysis provides a basis for direct comparison between design-basis analysis results with and without the release of tritium from TPBARs.

A discussion of the GENII code is provided in Appendix C. A general discussion of the MACCS2 computer code is provided in Section D.1.2.2. A detailed description of the MACCS model is provided in NUREG/CR-4691 (NRC 1990a). The enhancements incorporated in MACCS2 are described in the MACCS2 User's Guide (SNL 1997).

D.1.2.2 MACCS2 Computer Code

The MACCS2 computer code, Version 1.12, is used to estimate the radiological doses and health effects that could result from postulated accidental releases of radioactive materials to the atmosphere. The specification of the release characteristics, designated a "source term," can consist of up to four Gaussian plumes that are often referred to simply as "plumes."

The radioactive materials released are modeled as being dispersed in the atmosphere while being transported by the prevailing wind. During transport, whether or not there is precipitation, particulate material can be modeled as being deposited on the ground. If contamination levels exceed a user-specified criterion, mitigative actions can be triggered to limit radiation exposures.

There are two aspects of the code's structure that are basic to understanding its calculations: (1) the calculations are divided into modules and phases, and (2) the region surrounding the facility is divided into a polar-coordinate grid. These concepts are described in the following sections.

MACCS2 is divided into three primary modules: ATMOS, EARLY, and CHRONC. Three phases are defined as the emergency, intermediate, and long-term phases. The relationship among the code's three modules and the three phases of exposure are summarized below.

The ATMOS module performs all of the calculations pertaining to atmospheric transport, dispersion, and deposition, as well as the radioactive decay that occurs before release and while the material is in the atmosphere. It utilizes a Gaussian plume model with Pasquill-Gifford dispersion parameters. The phenomena treated include building wake effects, buoyant plume rise, plume dispersion during transport, wet and dry deposition, and radioactive decay and ingrowth. The results of the calculations are stored for use by EARLY and CHRONC. In addition to the air and ground concentrations, ATMOS stores information on wind direction, arrival and departure times, and plume dimensions.

The EARLY module models the time period immediately following a radioactive release. This period is commonly referred to as the emergency phase. The emergency phase begins at each successive downwind distance point when the first plume of the release arrives. The duration of the emergency phase is specified by the user, and it can range between one and seven days. The exposure pathways considered during this period are direct external exposure to radioactive material in the plume (cloudshine), exposure from inhalation of radionuclides in the cloud (cloud inhalation), exposure to radioactive material deposited on the ground (groundshine), inhalation of resuspended material (resuspension inhalation), and skin dose from material deposited on the skin. Mitigative actions that can be specified for the emergency phase include evacuation, sheltering, and dose-dependent relocation.

The CHRONC module performs all of the calculations pertaining to the intermediate and long-term phases. CHRONC calculates the individual health effects that result from both direct exposure to contaminated ground and from inhalation of resuspended materials, as well as indirect health effects caused by the consumption of contaminated food and water by individuals who could reside both on and off of the computational grid.

The intermediate phase begins at each successive downwind distance point upon the conclusion of the emergency phase. The user can configure the calculations with an intermediate phase that has a duration as short as zero or as long as one year. Essentially, there is no intermediate phase and a long-term phase begins immediately upon conclusion of the emergency phase.

These models are implemented on the assumption that the radioactive plume has passed and the only exposure sources (groundshine and resuspension inhalation) are from ground-deposited material. It is for this reason that MACCS2 requires the total duration of a radioactive release be limited to no more than four days. Potential doses from food and water ingestion during this period are not considered.

The mitigative action model for the intermediate phase is very simple. If the intermediate phase dose criterion is satisfied, the resident population is assumed to be present and subject to radiation exposure from groundshine and resuspension for the entire intermediate phase. If the intermediate phase exposure exceeds the dose criterion, then the population is assumed to be relocated to uncontaminated areas for the entire intermediate phase.

The long-term phase begins at each successive downwind distance point upon the conclusion of the intermediate phase. The exposure pathways considered during this period are groundshine, resuspension inhalation, and food and water ingestion.

The exposure pathways considered are those resulting from ground-deposited material. A number of protective measures can be modeled in the long-term phase to reduce doses to user-specified levels such as decontamination, temporary interdiction, and condemnation. The decisions on mitigative action in the long-term phase are based on two sets of independent actions: (1) decisions relating to whether land at a specific location and time is suitable for human habitation (habitability), and (2) decisions relating to whether land at a specific location and time is suitable for agricultural production (farmability).

All of the calculations of MACCS2 are stored on the basis of a polar-coordinate spatial grid with a treatment that differs somewhat between calculations of the emergency phase and calculations of the intermediate and long-term phases. The region potentially affected by a release is represented with an (r,θ) grid system centered on the location of the release. The radius, r, represents downwind distance. The angle, θ, is the angular offset from north, going clockwise.

The user specifies the number of radial divisions as well as their endpoint distances. The angular divisions used to define the spatial grid are fixed in the code and correspond to the 16 points of the compass, each being 22.5 degrees wide. The 16 points of the compass are used in the U.S. to express wind direction. The compass sectors are referred to as the coarse grid.

Since emergency phase calculations use dose-response models for early fatalities and early injuries that can be highly nonlinear, these calculations are performed on a finer grid basis than the calculations of the intermediate and long-term phases. For this reason, the calculations of the emergency phase are performed with the 16 compass sectors divided into three, five, or seven equal, angular subdivisions. The subdivided compass sectors are referred to as the fine grid.

The compass sectors are not subdivided into fine subdivisions for the intermediate and long-term phases because these calculations do not include estimation of the often highly nonlinear early fatality and early injury health effects, being limited to cancer and genetic effects. In contrast to the emergency phase, the calculations for these phases are performed using doses averaged over the full 22.5 degree compass sectors of the coarse grid.

Two types of doses may be calculated by the code: "acute" and "lifetime."

Acute doses are calculated to estimate deterministic health effects that can result from high doses delivered at high dose rates. Such conditions may occur in the immediate vicinity of a nuclear power plant following hypothetical severe accidents where containment failure has been assumed to occur. Examples of the health effects based on acute doses are early fatality, prodromal vomiting, and hypothyroidism.

Lifetime doses are the conventional measure of detriment used for radiological protection. These are 50-year dose commitments to either specific tissues (e.g., red marrow and lungs) or a weighted sum of tissue doses defined by the International Commission on Radiological Protection and referred to as "effective dose." Lifetime doses may be used to calculate the stochastic health effect risk resulting from exposure to radiation. MACCS2 uses the calculated lifetime dose in cancer risk calculations.

D.1.2.3 Data and General Assumptions

To assess the consequences of the accidents, with the exception of the beyond design-basis accidents, data were collected and produced and assumptions were made for incorporation in the GENII analyses. The source terms for the various accidents are described in Section D.1.1. The meteorological and population data are identical to those described in Appendix C. Ingestion parameters are based on Regulatory Guide 1.109 (NRC 1977).

To assess the consequences of beyond design-basis accidents, the following data and assumptions were incorporated into the MACCS2 analysis.

- The **nuclide inventory** at accident initiation (e.g., reactor trip) of those radioactive nuclides important for the calculation of offsite consequences for each reactor is given in Section D.1.1.

- The **atmospheric source term** produced by the accident is described by the number of plume segments released; sensible heat content; timing; duration; height of release for each plume segment; time when offsite officials are warned that an emergency response should be initiated; and for each important radionuclide, the fraction of that radionuclide's inventory released with each plume segment. The source terms for each accident scenario are provided in Section D.1.1.

- **Meteorological data** characteristics of the site region are described by one year of hourly windspeed, atmospheric stability, and rainfall recorded at each site. Although one year of hourly readings contains 8,760 weather sequences, MACCS2 calculations examine only a representative subset of these sequences. The representative subset is selected by sampling the weather sequences after sorting them into weather bins defined by windspeed, atmospheric stability, and intensity and distance of the occurrence of rain.

- The **population distribution information** about each reactor site is based on the 1990 U.S. Census of Population and Housing (DOC 1992). State and county population estimates were examined to extrapolate the 1990 data to the year 2025. This data was fitted to a polar coordinate grid with 16 angular sectors aligned with the 16 compass directions and 29 radial intervals that extend outward to 80 kilometers (50 miles).

- **Habitable land fractions** for the region around each reactor site were determined in a manner similar to the population distribution. The census block group boundary files include polygons that are classified as water features. The percentage of each sector that is covered by water is determined by fitting this data to the polar coordinate grid.

- **Farmland fractions** are the percentage of land devoted to farming (DOC 1993).

- **Emergency response assumptions** for evacuation, including delay time before evacuation, area evacuated, average evacuation speed, and travel distance, are provided in the Tennessee Multi-Jurisdictional Plans. Average evacuation speeds are based on the most conservative general population evacuation times.

- **Shielding and exposure data** must be input to the MACCS2 code. The code requires shielding factors be specified for people evacuating in vehicles (cars, buses); taking shelter in structures (houses, offices, schools); and continuing normal activities either outdoors, in vehicles, or indoors. Because inhalation doses depend on breathing rate, breathing rates must be specified for people who are continuing normal activities, taking shelter, and evacuating. Since indoor concentrations of gas-borne radioactive materials are usually substantially less than outdoor concentrations, MACCS2 also requires that inhalation and skin protection shielding factors (indoor/outdoor concentration ratios) be provided.

  The protection factors presented in **Table D–15** were used in the analyses. The values in Table D–15 are for the Sequoyah Nuclear Plant as stated in NUREG/CR-4551, and were used in the analysis for all three plants.

Table D–15 NUREG/CR–4551 Protection Factors

Protection Factor[a]	*Evacuees*	*Sheltering*	*Normal Activities*
Cloud Shielding Factor	1.0	0.65	0.75
Skin Protection Factor	1.0	0.33	0.41
Inhalation Protection Factor	1.0	0.33	0.41

[a] A protection factor of 1.0 indicates no protection, while a protection factor of 0.0 indicates 100 percent protection.

  For this analysis, the evacuation and sheltering region is defined as a 10-mile radial distance centered on the plant. A sheltering period is defined as the phase occurring before the initiation of the evacuation. During the sheltering phase, shielding factors appropriate for sheltered activity are used to calculate doses for the individuals in contaminated areas.

  At the end of the sheltering phase, the resident individuals begin their travel out of the region. Travel speeds and delay times are based on the Tennessee Multi-Jurisdictional Plans. The general population evacuation times for the various areas within the 10-mile radius are averaged to determine an overall evacuation delay time and evacuation speed for the Watts Bar and Sequoyah Nuclear Plants. Bellefonte Nuclear Plant evacuation plans were unavailable, so the Bellefonte evacuation parameters were based on the Sequoyah Nuclear Plant data.

- Maximally Exposed Offsite Individual Dose is the total dose estimated to be incurred by a hypothetical individual assumed to reside at a particular location on the spatial grid. Population data, therefore, have no bearing on the generation of this consequence measure. Only direct exposure is considered in these results. Exposures from the ingestion of contaminated food and water are not included. Also, the generation of these results takes full account of any mitigative action models activated by exceeding the dose thresholds. During evacuation, individuals have no protection from direct exposure. Therefore, in certain scenarios, it is possible that an evacuee may incur a larger direct exposure dose than an individual who does not evacuate.

- Long-term protective measures such as decontamination, temporary relocation, contaminated crops, milk condemnation, and farmland production prohibition are based on U.S. Environmental Protection Agency (EPA) Protective Action Guides.

- Mitigative actions (relocation, evacuation, interdiction, condemnation) are implemented for beyond design-basis accidents (vessel breach with containment bypass, vessel breach with early containment failure, and vessel breach with late containment failure).

- Dose conversion factors required by MACCS2 for the calculation of committed effective dose equivalents are cloudshine dose-rate factor; groundshine dose-rate factor; "lifetime" 50-year committed inhalation dose, used for calculation of individual and societal doses and stochastic health effects; and 50-year committed ingestion dose, used for calculation of individual and societal doses and stochastic health effects from food and water ingestion.

The MACCS2 dose conversion factor preprocessor FGRDCF was used to create the dose factors. FGRDCF incorporates the data of Federal Guidance Reports 11 and 12 (EPA 1988, EPA 1993). The inhalation and ingestion dose conversion factors are for the most part identical to the values listed in International Commission on Radiological Protection 30 (ICRP 1980). Revised metabolic models for the following transuranic elements: niobium, plutonium, americium, curium, berkelium, californium, einsteinium, fermium, and mendelevium are used (ICRP 1986). In addition, Federal Guidance Report 11 provides inhalation and ingestion dose conversion factors for a few radionuclides (strontium-82, technetium-95, technetium-95m, antimony-116, plutonium-246, and curium-250) not considered in International Commission on Radiological Protection 30, but for which nuclear decay data were presented in International Commission on Radiological Protection 38 (ICRP 1983). Federal Guidance Report 12 provides external dose-rate factors for the 825 nuclides identified in International Commission on Radiological Protection 38.

The only change made to the dose conversion factors produced by FGRDCF was to the tritium inhalation factor. The 50-year committed inhalation dose for tritium was increased by 50 percent to account for skin absorption (PNL 1988).

D.1.2.4 Health Effects Calculations

The following sections describe the technical approach used to calculate potential consequences to human health from exposure to radionuclides.

The health consequences from exposure to radionuclides from accidental releases were calculated. Total effective dose equivalents were calculated and converted to estimates of cancer fatalities using dose conversion factors recommended by the International Commission on Radiological Protection. For individuals, the estimated probability of a latent cancer fatality occurring is reported for the maximally exposed individual, an average individual in the population within 80 kilometers (50 miles), and a noninvolved worker.

The nominal values of lifetime cancer risk for low dose or low dose rate exposure (less than 20 rad) used in this EIS are 0.0005 per person-rem for a population of all ages and 0.0004 per person-rem for a working population. These dose-to-risk conversion factors are established by the National Council on Radiation

Protection and Measurement (NCRP 1993). See Appendix C for more detail regarding human health risk factors for nonfatal cancers and genetic disorders.

GENII uses a straight line plume method for calculating doses to receptors. The release/plume is assumed to disperse outward from the release point in one direction. Plume dispersion refers to the plume spreading out over a larger area and becoming less concentrated, which leads to lower doses. Certain weather conditions are better for plume dispersion than others. Therefore, it is necessary to analyze the doses to each receptor (e.g., the maximally exposed individual population and the noninvolved worker) for the 16 compass sectors at each site to determine the maximum sector doses. This maximum receptor dose is presented in this EIS. This analysis conservatively assumes that after the accident, the wind would blow towards the sector which produces maximum dosage. In addition, the GENII analyses assume that the accident occurs in autumn, which maximizes the estimated dose from contaminated food ingestion. Doses to each receptor were calculated using 50 percent meteorology. Fifty percent weather indicates a distribution with median weather conditions, (half of the weather conditions are worse and half are better). This meteorology is consistent with the guidance provided in the NRC's Regulatory Guide 4.2 (NRC 1976).

The MACCS2 code was applied in a probabilistic manner using a weather bin sampling technique. The weather bin sampling method sorts weather sequences into categories and assigns a probability to each category according to the initial conditions (wind speed and stability class) and the occurrence of rain. Each of the sampled meteorological sequences was applied to each of the 16 sectors (accounting for the frequency of occurrence of the wind blowing in that direction). Individual doses as a function of distance and direction were calculated for each of the meteorological sequence samples. The mean dose values of the sequences were generated for each of the 16 sectors. The highest of these dose values was used for the maximally exposed individual and the noninvolved worker. Population doses are the sum of the individual doses in each sector.

D.1.2.5 Deterministic Calculations

D.1.2.5.1 Introduction

In addition to the GENII and MACCS2 calculations, deterministic analyses were performed for the reactor and nonreactor design-basis accidents (large break loss-of-coolant accident and waste gas decay tank rupture). The deterministic analyses were performed to provide a comparison of the effect of tritium on the doses calculated in the candidate reactor Final Safety Analysis Reports. The Final Safety Analysis Reports present the thyroid inhalation, whole body beta, and whole body gamma doses at the exclusion area boundary and the low population zone. The deterministic analyses calculate the additional dose attributable to tritium using the same method as the Final Safety Analysis Reports.

D.1.2.5.2 Large Break Loss-of-Coolant Accident

To determine the effects of a tritium release following a postulated design-basis accident, a deterministic analysis based on Regulatory Guide 1.4 (NRC 1974) was adopted. The Regulatory Guide 1.4 analysis was incorporated in the candidate reactor Safety Analysis Reports to calculate the environmental effects resulting from a design-basis large break loss-of-coolant accident event. The following paragraphs describe the release paths from containment to the environment, the conservatisms employed, and the dose calculation method.

The primary containment leak rate used in the Final Safety Analysis Report analyses for the first 24 hours is the design-basis leak rate (as specified in the technical specifications regarding containment leakage), and it is 50 percent of this value for the duration of the accident. The Watts Bar and Sequoyah Final Safety Analysis Reports assume the primary containment (known here as steel containment vessel) leak rates to be 0.25 percent of the containment atmosphere per day for the first 24 hours following the accident and 0.125 percent per day for the remainder of the 30-day period. The Bellefonte Final Safety Analysis Report assumes the leak rate to be 0.2 percent per day for the first 24 hours following the accident and 0.1 percent per day for the remainder of the 30-day period.

For the Watts Bar and Sequoyah Nuclear Plants, the leakage from the steel containment vessel can be grouped into two categories: leakage into the auxiliary building and leakage into the annulus (a space between the steel containment vessel and shield building where leakage from primary containment is collected before it is released). For the Bellefonte Nuclear Plant, the leakage from the primary containment can be grouped into three categories: leakage into the auxiliary building, leakage into the annulus (a space between primary and secondary containment), and leakage directly to the environment.

The Watts Bar and Sequoyah Nuclear Plant analyses assume that 25 percent of the total primary leakage goes to the auxiliary buildings. This value is an estimated upper bound of leakage to the auxiliary buildings based on 10 CFR 50, Appendix J, testing of all containment penetrations. Selecting an upper bound is conservative because an increased leakage fraction to the auxiliary building would result in an increased offsite dose. The Bellefonte Nuclear Plant analysis assumes that 9.5 percent of the total primary leakage goes to the auxiliary building.

At the Watts Bar and Sequoyah Nuclear Plants, the auxiliary building is normally ventilated by the auxiliary building ventilation system. However, following a large break loss-of-coolant accident, the normal ventilation systems to all areas of the auxiliary building would be shut down and isolated. Upon auxiliary building isolation, the auxiliary building gas treatment system would be activated to ventilate the area and filter the exhaust to the atmosphere. At the Bellefonte Nuclear Plant, during both normal and emergency operations, the auxiliary building's engineered safety feature environmental control system provides pressure control and cleanup.

At each plant, fission products that leak from the primary containment to areas of the auxiliary building would be diluted in the room atmosphere and would travel through ducts and other rooms to the areas where the suctions for the auxiliary building gas treatment system or environmental control system are located. The Final Safety Analysis Report analyses allow a holdup time for airborne activity after an initial period of direct release. However, for the tritium analysis, it is conservatively assumed that activity leaking to the auxiliary building would be released directly to the environment through the auxiliary building gas treatment system or environmental control system, neglecting any holdup time in the auxiliary building before being exhausted.

The Watts Bar and Sequoyah Nuclear Plant analyses assume that 75 percent of the primary containment leakage would be to the annulus (TVA 1995a, TVA 1996). The Bellefonte Nuclear Plant analysis assumes that 90 percent of the primary containment leakage would be to the annulus (TVA 1991). The presence of the annulus between the primary containment (or steel containment vessel) and the secondary containment (or shield building) reduces the probability of direct leakage from the containment to the atmosphere and allows holdup and plate-out of fission products in the shield building. For the tritium analysis, plate-out in the annulus is neglected.

Transfer of activity from the annulus volume to the emergency gas treatment system suction for the Watts Bar and Sequoyah Nuclear Plants, or to the secondary containment cleanup system suction for the Bellefonte Nuclear Plant, is assumed to be a statistical process mathematically similar to the decay process (i.e., the rate of removal from the annulus is proportional to the activity in the annulus). This corresponds to an assumption that the activity is homogeneously distributed throughout the mixing volume. Because of the low emergency gas treatment system or secondary containment cleanup system flow rate compared to the annulus volume, the thermal convection due to heating of the containment structure, and the relative location of the emergency gas treatment system or secondary containment cleanup system suctions and the emergency gas treatment system or secondary containment cleanup system recirculation exhausts, a high degree of mixing can be expected.

It is, however, conservatively assumed that only 50 percent of the annulus free volume is available for mixing of the activity.

The emergency gas treatment system and secondary containment cleanup system are essentially annulus recirculation systems with pressure-activated valves that allow part of the system flow to be exhausted to the atmosphere to maintain an adequate annulus pressure. It is conservatively assumed that, for the first hour following the accident, all of the available tritium is exhausted. The holdup time is a function of the emergency gas treatment system or secondary containment cleanup system flow and exhaust rates, as well as the annulus volume. The holdup time before release is defined as 50 percent of the annulus volume divided by the exhaust flow rate of the emergency gas treatment system or secondary containment cleanup system.

The annulus pressure would be maintained at less than the auxiliary building's internal pressure during normal operation; therefore, any leakage between the two volumes following a loss-of-coolant accident would be into the annulus. It is conservatively assumed that there is no leakage via this route.

The Bellefonte Nuclear Plant also has a leakage of 0.5 percent of the total primary containment leak rate directly to the environment. This leakage is assumed to pass directly to the environment without mixing or holdup.

In the Final Safety Analysis Reports, thyroid inhalation and external whole body gamma and beta doses are calculated at the exclusion area boundary and low population zone. The inhalation and beta doses for tritium are calculated; no gamma dose calculation is needed since tritium decays only by beta emission.

The exclusion area boundary is that area surrounding the reactor in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility that they interfere with normal operations of the facility and appropriate and effective arrangements are made to control traffic and protect public health and safety on the highway, railroad, or waterway in an emergency. Residences within the exclusion area normally would be prohibited. In any event, residents would be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety would result.

The low population zone is the area immediately surrounding the exclusion area that contains residents whose total number and density indicate there is a reasonable probability that appropriate protective measures could be taken on their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. For example, whether a specific number of people can be evacuated from a specific area or instructed to take shelter on a timely basis would depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.

Calculations are performed using hourly time steps. This time step size is appropriate because of the large primary containment volume and low leakage rate; the tritium concentration (activity per volume) decreases only a few tenths of a percent per hour. At each time step the activity per hour is calculated and placed in the thyroid inhalation and beta dose formulas shown below to determine the doses. Final Safety Analysis Report time-dependent atmospheric dispersion factors, breathing rates, and dose conversion factors are incorporated. The doses at each time step are summed for a total dose. Doses are calculated separately for each pathway (annulus, auxiliary building, bypass), and then summed.

Thyroid inhalation doses are calculated using the following equation (NRC 1974, AEC 1972).

$$Dose = \left(\frac{X}{Q}\right)_t \cdot BR_t \cdot Q_t \cdot DCF$$

where:

$\left(\frac{X}{Q}\right)_t$	is the average atmospheric dilution factor over a given time interval t
Br_t	is the breathing rate for time interval t
Q_t	is the activity of tritium released during a given time interval t
DCF	is the inhalation dose conversion factor for tritium

Whole body beta doses are calculated using the following equation (NRC 1974, AEC 1972).

$$Dose = 0.23 \cdot \left(\frac{X}{Q}\right)_t \cdot Q_t \cdot \overline{E_\beta}$$

where:

$\left(\frac{X}{Q}\right)_t$	is the average atmospheric dilution factor over a given time interval t
Q_t	is the activity of tritium released during a given time interval t
$\overline{E_\beta}$	is the average beta radiation energy emitted by tritium per disintegration

D.1.2.5.3 Waste Gas Decay Tank Accident

The effects of a tritium release following a postulated waste gas decay tank rupture also are analyzed using a deterministic approach. As in the Final Safety Analysis Reports, this analysis is based on Regulatory Guide 1.24 (AEC 1972). The tritium source term available for release from the waste gas decay tank is described in Section D.1.1. The inventory of the waste gas decay tank is assumed to leak out at ground level over a two-hour time period. Thyroid inhalation and whole-body beta doses are calculated for the exclusion area boundary and the low population zone using the equations described in Section D.1.2.5.2. Final Safety Analysis Report time-dependent atmospheric dispersion factors, breathing rates, and dose conversion factors are incorporated.

D.1.2.6 Uncertainties

The sequence of analyses performed to generate the radiological and hazardous chemicals impacts estimates from normal operation of commercial light water reactor (CLWR) facilities, CLWR facility accidents, and overland transportation include: (1) selection of normal operational modes and accident scenarios and their probabilities, (2) estimation of source terms, (3) estimation of environmental transport and uptake of radionuclides and hazardous chemicals, (4) calculation of radiation and chemical doses to exposed individuals, and (5) estimation of health effects. Health effects are presented in terms of latent cancers and latent cancer fatalities. There are uncertainties associated with each of these steps. Uncertainties exist in the way the physical systems being analyzed are represented by the computational models and in the data required to exercise the models (due to measurement errors, sampling errors, or natural variability).

Of particular interest are the uncertainties in the estimates of cancer deaths from exposure to radioactive materials. The numerical values of the health risk estimates used in this EIS (refer to C.2.1.2) are obtained by the practice of linear extrapolation from the nominal risk estimate for lifetime total cancer mortality resulting from exposures at 10 rad. Other methods of extrapolation to the low-dose region could yield higher or lower estimates of cancer deaths. Studies of human populations exposed at low doses are inadequate to demonstrate the actual level of risk. There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiological observation, and the possibility of no risk or even health benefits (hormesis effects) cannot be excluded. Because the health risk estimators are multiplied by conservatively calculated radiological doses to predict fatal cancer risks, the fatal cancer values presented in this EIS are expected to be overestimates.

For the purposes of presentation in this EIS, the impacts calculated from the linear model are treated as an upper bound case, consistent with the widely used methodologies for quantifying radiogenic health impacts. This does not imply that health effects are expected. Moreover, in cases where the upper bound estimators predict a number of latent cancer deaths that is greater than one, this does not imply that the latent cancer deaths are identifiable to any individual.

Uncertainties are also introduced when accident analyses performed for similar existing facilities have been used as a major source of data. Although the radionuclide composition of source terms are reasonable estimates, there are uncertainties in the radionuclide inventory and release fractions that affect the estimated consequences. Accident frequencies for low probability sequences of events are always difficult to estimate, even for operating facilities, because there is little or no record of historical occurrences. For a new facility, such as Bellefonte 1 or 2, any use of accident frequencies that are estimated from similar exiting facilities would tend to further compound the effects of uncertainties.

In summary, the radiological and hazardous chemical impact estimates presented in this EIS were obtained by:

- Using the latest available data
- Considering the processes, events, and accidents reasonably foreseeable for tritium production in a CLWR and overland transportation of irradiated TPBARs
- Making conservative assumptions when there is doubt about the exact nature of the processes and events taking place, such that the chance of underestimating health impacts is small

D.1.3 Accident Consequences and Risks

D.1.3.1 Reactor Design-Basis Accident

The reactor design-basis accident source term and accident frequency data, presented in Tables D–2 and D–3, were evaluated using two different accident analysis approaches. The first analysis approach used the GENII accident analysis computer code (PNL 1988) to estimate the accident consequences and risks. The second analysis approach was based on published NRC guidance for the assessment of design-basis accident impacts. The NRC requires that the results of an analysis evaluating design-basis accident impacts on a different set of receptors be submitted for evaluation as part of the licensing basis for each reactor.

Analyses were performed in accordance with guidance provided in NRC Regulatory Guide 4.2 (NRC 1976). This guide recommends using an atmospheric diffusion value (χ/Q value) corresponding to 1/10 of the value determined in Safety Guide No. 4. This safety guide has been revised and reissued as Revision 2, Regulatory Guide 1.4 (NRC 1974). The NRC in 1983 issued Regulatory Guide 1.145, providing guidance in determining 95th percentile χ/Q values using a site meteorological direction-dependent approach (NRC 1983). In these analyses, DOE assumes the 95 percentile direction-dependent χ/Q values are consistent with the guidance provided in Safety Guide No. 4 and Regulatory Guide 1.4. The GENII computer code, which is based on the current NRC's acceptable directional dependent approach, was used to determine 50 percentile and 95 percentile meteorological conditions for each site. The results indicated that the estimated doses using 50 percentile meteorological conditions were more than 0.1 times the 95 percentile meteorological doses. Therefore, the 50 percentile meteorological condition at each site was used to estimate the consequences of design-basis and TPBAR handling accidents.

Table D–16 summarizes the GENII-generated consequences of the reactor design-basis accident to the maximally exposed offsite individual, an average individual in the public within an 80-kilometer (50-mile) radius of the reactor site, a noninvolved worker at the Watts Bar and Bellefonte Nuclear Plant Sites located 640 meters (0.4 miles) from the release point, and a noninvolved worker at the Sequoyah Nuclear Plant located at the site boundary 556 meters (0.35 miles) from the release point. The risks associated with the reactor design-basis accident to the same receptors are summarized in **Table D–17**.

Table D–18 summarizes the consequences of the reactor design-basis accident (estimated using NRC guidance and 95th percentile χ/Q values) to an individual located at the reactor site exclusion area boundary and an individual located at the reactor site low population zone. The 0 TPBAR entries represent total accident dose compared to the 1,000 and 3,400 TPBAR entries, which represent the incremental change to the dose due to the addition of TPBARs. The margin-to-site dose limits (i.e., the difference between the dose estimate and the site dose criteria) associated with the reactor design-basis accident to the same receptors are summarized in **Table D–19**.

D.1.3.2 Nonreactor Design-Basis Accident

The nonreactor design-basis accident source term and accident frequency data presented in Section D.1.1.3 were evaluated using two different accident analysis approaches. The first analysis approach used the GENII accident analysis computer code (PNL 1988) to estimate the accident consequences and risks. The second analysis approach was based on published NRC guidance for the assessment of design-basis accident impacts. The NRC requires that the results of an analysis evaluating design-basis accident impacts on a different set of receptors be submitted for evaluation as part of the licensing basis for each reactor.

Table D–16 GENII-Generated Reactor Design-Basis Accident Consequences

Reactor Site	*Tritium Production*	*Maximally Exposed Offsite Individual*		*Average Individual in Population to 80 kilometers (50 miles)*		*Noninvolved Worker*	
		Dose (rem)	*Cancer Fatality* [a]	*Dose (rem)*	*Cancer Fatality* [a]	*Dose (rem)*	*Cancer Fatality* [a]
Watts Bar	1,000 TPBARs	0.0014	7.0×10^{-7}	0.000011	5.5×10^{-9}	0.000024	9.6×10^{-9}
	3,400 TPBARs	0.0047	2.4×10^{-6}	0.000038	1.9×10^{-8}	0.000081	3.2×10^{-8}
Sequoyah	1,000 TPBARs	0.0019	9.5×10^{-7}	0.000022	1.1×10^{-8}	8.1×10^{-6}	3.2×10^{-9}
	3,400 TPBARs	0.0065	3.3×10^{-6}	0.000075	3.8×10^{-8}	0.000028	1.1×10^{-8}
Bellefonte	1,000 TPBARs	0.000085	4.3×10^{-8}	1.7×10^{-6}	8.5×10^{-10}	2.9×10^{-8}	1.2×10^{-11}
	3,400 TPBARs	0.00029	1.5×10^{-7}	5.5×10^{-6}	2.8×10^{-9}	1.0×10^{-7}	4.0×10^{-11}

[a] Increased likelihood of cancer fatality.

Table D–17 Reactor Design-Basis Accident Annual Risks

Reactor Site	*Tritium Production*	*Maximally Exposed Offsite Individual* [a]	*Average Individual in Population to 80 kilometers (50 miles)* [a]	*Noninvolved Worker* [a]
Watts Bar	1,000 TPBARs	1.4×10^{-10}	1.1×10^{-12}	1.9×10^{-12}
	3,400 TPBARs	4.8×10^{-10}	3.8×10^{-12}	6.4×10^{-12}
Sequoyah	1,000 TPBARs	1.9×10^{-10}	2.2×10^{-12}	6.4×10^{-13}
	3,400 TPBARs	6.6×10^{-10}	7.6×10^{-12}	2.2×10^{-12}
Bellefonte	1,000 TPBARs	8.6×10^{-12}	1.7×10^{-13}	2.4×10^{-15}
	3,400 TPBARs	3.0×10^{-11}	5.6×10^{-13}	8.0×10^{-15}

[a] Increased likelihood of cancer fatality per year.

Table D–18 Reactor Design-Basis Accident Consequences Using the NRC Analysis Approach

Reactor Site	*Tritium Production*	*Dose Description*	*Individual at Area Exclusion Boundary Dose (rem)*	*Individual at Low Population Zone Dose (rem)*
Watts Bar	0 TPBARs (No Action) [a]	Thyroid Inhalation Dose	34.1	11.0
		Beta + Gamma Whole Body Dose	3.5	3.4
	1,000 TPBARs [b]	Thyroid Inhalation Dose	0.0018	0.0022
		Beta + Gamma Whole Body Dose	0.00010	0.00018
	3,400 TPBARs [b]	Thyroid Inhalation Dose	0.0060	0.0075
		Beta + Gamma Whole Body Dose	0.00035	0.00061
Sequoyah	0 TPBARs (No Action) [a]	Thyroid Inhalation Dose	145	27
		Beta + Gamma Whole Body Dose	12.2	2.9
	1,000 TPBARs [b]	Thyroid Inhalation Dose	0.0044	0.0018
		Beta + Gamma Whole Body Dose	0.00026	0.0001
	3,400 TPBARs [b]	Thyroid Inhalation Dose	0.015	0.0060
		Beta + Gamma Whole Body Dose	0.00088	0.00047

Reactor Site	Tritium Production	Dose Description	Individual at Area Exclusion Boundary Dose (rem)	Individual at Low Population Zone Dose (rem)
Bellefonte	0 TPBARs [c, d]	Thyroid Inhalation Dose	5.8	2.7
		Beta + Gamma Whole Body Dose	0.031	0.18
	1,000 TPBARs [b]	Thyroid Inhalation Dose	0.0041	0.0028
		Beta + Gamma Whole Body Dose	0.00024	0.00021
	3,400 TPBARs [b]	Thyroid Inhalation Dose	0.011	0.0095
		Beta + Gamma Whole Body Dose	0.00082	0.00073

[a] TVA 1995a, TVA 1996.
[b] Only TPBAR contribution to dose.
[c] TVA 1991.
[d] The 0 TPBAR entry is included for consistency with the Watts Bar and Sequoyah Nuclear Plant analyses. The No Action alternative at the Bellefonte Nuclear Plant implies that the reactors are not brought into commercial service. The No Action radiological dose is 0.

Table D–19 Reactor Design-Basis Accident Consequence Margin to Site Dose Criteria

Reactor Site	Tritium Production	Dose Description [a]	Site Dose Criteria (rem) [b]	Individual at Area Exclusion Boundary		Individual at Low Population Zone	
				Dose (rem)	Margin (%) [c]	Dose (rem)	Margin (%) [c]
Watts Bar	0 TPBARs (No Action) [d]	Thyroid Inhalation Dose	300	34.1	88.6	11.0	96.3
		Beta + Gamma Whole Body Dose	25	3.5	86.1	3.4	86.2
	1,000 TPBARs	Thyroid Inhalation Dose	300	34.1	88.6	11.0	96.3
		Beta + Gamma Whole Body Dose	25	3.5	86.1	3.4	86.2
	3,400 TPBARs	Thyroid Inhalation Dose	300	34.1	88.6	11.0	96.3
Sequoyah	0 TPBARs	Beta + Gamma Whole Body Dose	25	3.5	86.1	3.4	86.2
	(No Action) [d]	Thyroid Inhalation Dose	300	145	51.6	27	91.0
	1,000 TPBARs	Beta + Gamma Whole Body Dose	25	12.2	51.1	2.9	88.4
		Thyroid Inhalation Dose	300	145	51.6	27	91.0
		Beta + Gamma Whole Body Dose	25	12.2	51.1	2.9	88.4
	3,400 TPBARs	Thyroid Inhalation Dose	300	145	51.6	27	91.0
		Beta + Gamma Whole Body Dose	25	12.2	51.1	2.9	88.4
Bellefonte	0 TPBARs [e, f]	Thyroid Inhalation Dose	300	5.8	98.1	2.7	99.1
		Beta + Gamma Whole Body Dose	25	0.031	99.9	0.18	99.3
	1,000 TPBARs	Thyroid Inhalation Dose	300	5.8	98.1	2.7	99.1
		Beta + Gamma Whole Body Dose	25	0.031	99.9	0.18	99.3
	3,400 TPBARs	Thyroid Inhalation Dose	300	5.9	98.0	2.7	99.1
		Beta + Gamma Whole Body Dose	25	0.032	99.9	0.18	99.3

[a] Dose is the total dose from the reactor plus the contribution from the TPBARs.
[b] 10 CFR 100.11.
[c] Margin below the site dose criteria.
[d] TVA 1995a, TVA 1996.
[e] TVA 1991.
[f] The 0 TPBAR entry is included for consistency with the Watts Bar and Sequoyah Nuclear Plant analyses. The No Action Alternative at the Bellefonte Nuclear Plant implies that the reactors are not brought into commercial service. The No Action Alternative radiological dose is 0.

Analyses were performed in accordance with guidance provided in NRC Regulatory Guide 4.2 (NRC 1976). **Table D–20** summarizes the GENII-generated consequences of the nonreactor design-basis accident with 50 percent meteorological conditions to the maximally exposed offsite individual, an average individual within an 80-kilometer (50-mile) radius of the reactor site, a noninvolved worker at the Watts Bar and Bellefonte Nuclear Plant sites located 640 meters (0.4 miles) from the release point, and a noninvolved worker at the Sequoyah Nuclear Plant located at the site boundary 556 meters (0.35 miles) from the release point. The risks associated with the nonreactor design-basis accident to the same receptors are summarized in **Table D–21.**

Table D–22 summarizes the consequences of the nonreactor design-basis accident to an individual located at the reactor site exclusion area boundary and an individual located at the reactor site low population zone. NRC guidance was used to derive these estimates. The 0 TPBAR entries represent total accident dose as opposed to the 1,000 and 3,400 TPBAR entries, which represent the incremental change to the dose due to the addition of TPBARs. The margin to NRC dose limits (i.e., the difference between the dose estimate and the site dose limit) associated with the reactor design-basis accident to the same receptors are summarized in **Table–23.**

Table D–20 GENII-Generated Nonreactor Design-Basis Accident Consequences

		Maximally Exposed Offsite Individual		*Average Individual in Population to 80 kilometers (50 miles)*		*Noninvolved Worker*	
Reactor Site	***Tritium Production***	***Dose (rem)***	***Cancer Fatality[a]***	***Dose (rem)***	***Cancer Fatality[a]***	***Dose (rem)***	***Cancer Fatality[a]***
Watts Bar	1,000 TPBARs	0.0067	3.4×10^{-6}	0.000079	4.0×10^{-8}	0.00010	4.2×10^{-8}
	3,400 TPBARs	0.022	0.000011	0.00027	1.4×10^{-7}	0.00036	1.5×10^{-7}
Sequoyah	1,000 TPBARs	0.0016	7.9×10^{-7}	0.00012	6.1×10^{-8}	0.000032	1.3×10^{-8}
	3,400 TPBARs	0.0054	2.7×10^{-6}	0.00042	2.1×10^{-7}	0.00011	4.5×10^{-8}
Bellefonte	1,000 TPBARs	0.00016	7.9×10^{-8}	0.000043	2.2×10^{-8}	3.1×10^{-7}	1.2×10^{-10}
	3,400 TPBARs	0.00054	2.7×10^{-7}	0.00015	7.4×10^{-8}	1.1×10^{-6}	4.3×10^{-10}

[a] Increased likelihood of cancer fatality.

Table D–21 Nonreactor Design-Basis Accident Annual Risks

Reactor Site	***Tritium Production***	***Maximally Exposed Offsite Individual[a]***	***Average Individual in Population to 80 kilometers (50 miles)[a]***	***Noninvolved Worker[a]***
Watts Bar	1,000 TPBARs	3.4×10^{-8}	4.0×10^{-10}	4.2×10^{-10}
	3,400 TPBARs	1.1×10^{-7}	1.4×10^{-9}	1.5×10^{-9}
Sequoyah	1,000 TPBARs	7.9×10^{-9}	6.1×10^{-10}	1.3×10^{-10}
	3,400 TPBARs	2.7×10^{-8}	2.1×10^{-9}	4.5×10^{-10}
Bellefonte	1,000 TPBARs	7.9×10^{-10}	2.2×10^{-10}	1.2×10^{-12}
	3,400 TPBARs	2.7×10^{-9}	7.4×10^{-10}	4.3×10^{-12}

[a] Increased likelihood of cancer fatality per year.

Table D–22 Nonreactor Design-Basis Accident Consequences Using the NRC Analysis Approach

Reactor Site	Tritium Production	Dose Description	Individual at Area Exclusion Boundary Dose (rem)	Individual at Low Population Zone Dose (rem)
Watts Bar	0 TPBARs (No Action) [a]	Thyroid Inhalation Dose	0.018	0.0042
		Beta + Gamma Whole Body Dose	0.13	0.031
	1,000 TPBARs [b]	Thyroid Inhalation Dose	0.0020	0.00048
		Beta + Gamma Whole Body Dose	0.00012	0.000028
	3,400 TPBARs [b]	Thyroid Inhalation Dose	0.0068	0.0016
		Beta + Gamma Whole Body Dose	0.00040	0.000097
Sequoyah	0 TPBARs (No Action) [a]	Thyroid Inhalation Dose	0.000013	1.1×10^{-6}
		Beta + Gamma Whole Body Dose	0.0017	0.00014
	1,000 TPBARs [b]	Thyroid Inhalation Dose	0.0055	0.00065
	3,400 TPBARs [b]	Beta + Gamma Whole Body Dose	0.00032	0.000039
		Thyroid Inhalation Dose	0.019	0.0022
		Beta + Gamma Whole Body Dose	0.0011	0.00013
Bellefonte	0 TPBARs [a, c]	Thyroid Inhalation Dose	0.0067	0.0019
		Beta + Gamma Whole Body Dose	0.71	0.14
	1,000 TPBARs [b]	Thyroid Inhalation Dose	0.0067	0.0013
		Beta + Gamma Whole Body Dose	0.00039	0.000079
	3,400 TPBARs [b]	Thyroid Inhalation Dose	0.023	0.0045
		Beta + Gamma Whole Body Dose	0.0013	0.00027

[a] TVA 1991, TVA 1995a, TVA 1996.
[b] Only TPBAR contribution to dose.
[c] The 0 TPBAR entry is included for consistency with the Watts Bar and Sequoyah Nuclear Plant analyses. The No Action Alternative at the Bellefonte Nuclear Plant implies that the reactors are not brought into commercial service. The No Action Alternative radiological dose is 0.

Table D–23 Nonreactor Design-Basis Accident Consequence Margin to Site Dose Criteria

Reactor Site	Tritium Production	Dose Description [a]	Site Dose Criteria (rem) [b]	Individual at Area Exclusion Boundary		Individual at Low Population Zone	
				Dose (rem)	Margin (%) [c]	Dose (rem)	Margin (%) [c]
Watts Bar	0 TPBARs (No Action) [d]	Thyroid Inhalation Dose	300	0.018	99.994	0.0042	99.999
		Beta + Gamma Whole Body Dose	25	0.13	99.5	0.031	99.9
	1,000 TPBARs	Thyroid Inhalation Dose	300	0.020	99.993	0.0047	99.998
		Beta + Gamma Whole Body Dose	25	0.13	99.5	0.031	99.9
	3,400 TPBARs	Thyroid Inhalation Dose	300	0.025	99.92	0.0058	99.998
		Beta + Gamma Whole Body Dose	25	0.13	99.5	0.031	99.9

Reactor Site	Tritium Production	Dose Description [a]	Site Dose Criteria (rem) [b]	Individual at Area Exclusion Boundary		Individual at Low Population Zone	
				Dose (rem)	Margin (%) [c]	Dose (rem)	Margin (%) [c]
Sequoyah	0 TPBARs (No Action) [d]	Thyroid Inhalation Dose	300	0.000013	100	1.1×10^{-6}	100
		Beta + Gamma Whole Body Dose	25	0.0017	99.993	0.00014	99.999
	1,000 TPBARs	Thyroid Inhalation Dose	300	0.0055	99.98	0.00065	99.999
		Beta + Gamma Whole Body Dose	25	0.0020	99.992	0.00018	99.999
	3,400 TPBARs	Thyroid Inhalation Dose	300	0.019	99.994	0.0022	99.999
		Beta + Gamma Whole Body Dose	25	0.0028	99.989	0.00027	99.998
Bellefonte	0 TPBARs [e, f]	Thyroid Inhalation Dose	300	0.0067	99.998	0.0019	99.99
		Beta + Gamma Whole Body Dose	25	0.71	97.2	0.14	99.4
	1,000 TPBARs	Thyroid Inhalation Dose	300	0.013	99.996	0.0032	99.999
		Beta + Gamma Whole Body Dose	25	0.71	97.2	0.14	99.4
	3,400 TPBARs	Thyroid Inhalation Dose	300	0.029	99.990	0.0064	99.998
		Beta + Gamma Whole Body Dose	25	0.71	97.2	0.14	99.4

[a] Dose is the total dose from the reactor plus the dose from the TPBARs.
[b] 10 CFR 100.11.
[c] Margin below the site dose criteria.
[d] TVA 1995a, TVA 1996.
[e] Bellefonte Final Safety Analysis Report (TVA 1991), realistic analysis dose estimates. Design analysis dose estimates were also below the site dose limits.
[f] The 0 TPBAR entry is included for consistency with the Watts Bar and Sequoyah Nuclear Plant analyses. The No Action Alternative at the Bellefonte Nuclear Plant implies that the reactors are not brought into commercial service. The No Action Alternative radiological dose is 0.

D.1.3.3 TPBAR Handling Accident

The TPBAR handling accident source term and accident frequency data presented in Section D.1.1.4 were evaluated using the GENII accident analysis computer code (PNL 1988). Analyses were performed in accordance with guidance provided in NRC Regulatory Guide 4.2 (NRC 1976). **Table D–24** summarizes the consequences of the TPBAR handling accident to the maximally exposed offsite individual, an average individual in the public within an 80-kilometer (50-mile) radius of the reactor site, a noninvolved worker at the Watts Bar and Bellefonte Nuclear Plant sites located 640 meters (0.4 miles) from the release point, and a noninvolved worker at the Sequoyah Nuclear Plant located at the site boundary 556 meters (0.35 miles) from the release point. The analysis assumes that no action would be taken on the site to reduce the dose to the noninvolved worker, and that the worker is exposed for 2,000 hours during the airborne release over the postulated one-year period. Calculations indicate that routine plant administrative controls and work permits for workers in the fuel pool area would require protective equipment (e.g., supplied air or air packs) and protective clothing for approximately one week after the accident due to the concentration of tritiated water vapor in the work area. The risks associated with the TPBAR handling accident to the same receptors are summarized in **Table D–25**.

Table D–24 TPBAR Handling Accident Consequences

Reactor Site	Maximally Exposed Offsite Individual		Average Individual in Population to 80 kilometers (50 miles)		Noninvolved Worker	
	Dose (rem)	Cancer Fatality [a]	Dose (rem)	Cancer Fatality [a]	Dose (rem)	Cancer Fatality [a]
Watts Bar	0.028	0.000014	0.00031	1.6×10^{-7}	0.0017	6.8×10^{-7}
Sequoyah	0.036	0.000018	0.00029	1.5×10^{-7}	0.0014	5.6×10^{-7}
Bellefonte	0.0045	2.3×10^{-6}	0.00025	1.3×10^{-7}	0.00007	2.8×10^{-8}

[a] Increased likelihood of cancer fatality.

Table D–25 TPBAR Handling Accident Annual Risks

Reactor Site	Tritium Production	Maximally Exposed Offsite Individual [a]	Average Individual in Population to 80 kilometers (50 miles) [b]	Noninvolved Worker [a]
Watts Bar	1,000 TPBARs	2.4×10^{-8}	2.7×10^{-10}	1.2×10^{-9}
	3,400 TPBARs	8.1×10^{-8}	9.3×10^{-10}	3.9×10^{-9}
Sequoyah	1,000 TPBARs	3.1×10^{-8}	2.6×10^{-10}	9.5×10^{-10}
	3,400 TPBARs	1.0×10^{-7}	8.7×10^{-10}	3.2×10^{-9}
Bellefonte	1,000 TPBARs	3.9×10^{-9}	2.2×10^{-10}	4.8×10^{-11}
	3,400 TPBARs	1.3×10^{-8}	7.5×10^{-10}	1.6×10^{-10}

[a] Increased likelihood of cancer fatality per year.

D.1.3.4 Truck Transportation Cask Handling Accident

The truck transportation cask handling accident source term and accident frequency data presented in Section D.1.1.5 were evaluated using the GENII accident analysis computer code (PNL 1988). Analyses were performed in accordance with guidance provided in NRC Regulatory Guide 4.2 (NRC 1976). **Table D–26** summarizes the consequences of the truck transportation cask handling accident to the maximally exposed offsite individual, an average individual in the public within an 80-kilometer (50-mile) radius of the reactor site, a noninvolved worker at the Watts Bar and Bellefonte Nuclear Plant sites located 640 meters (0.4 miles) from the release point, and a noninvolved worker at the Sequoyah Nuclear Plant located at the site boundary 556 meters (0.35 miles) from the release point. The analysis assumes that no action would be taken on site to reduce the dose to the noninvolved worker and that the worker is exposed for 2,000 hours during the airborne release over the postulated one-year period. The risks associated with the truck transportation cask handling accident to the same receptors are summarized in **Table D–27**.

Table D–26 Truck Transportation Cask Handling Accident Consequences

	Maximally Exposed Offsite Individual		*Average Individual in Population to 80 kilometers (50 miles)*		*Noninvolved Worker*	
Reactor Site	***Dose (rem)***	***Cancer Fatality*** [a]	***Dose (rem)***	***Cancer Fatality*** [a]	***Dose (rem)***	***Cancer Fatality*** [a]
Watts Bar	0.00072	3.6×10^{-7}	8.0×10^{-6}	4.0×10^{-9}	0.000043	1.7×10^{-8}
Sequoyah	0.00093	4.7×10^{-7}	7.5×10^{-6}	3.8×10^{-9}	0.000036	1.4×10^{-8}
Bellefonte	0.00012	6.0×10^{-8}	6.4×10^{-6}	3.2×10^{-9}	1.8×10^{-6}	7.2×10^{-10}

[a] Increased likelihood of cancer fatality.

Table D–27 Truck Transportation Cask Handling Accident Annual Risks

Reactor Site	***Tritium Production***	***Maximally Exposed Offsite Individual*** [a]	***Average Individual in Population to 80 kilometers (50 miles)*** [a]	***Noninvolved Worker*** [a]
Watts Bar	1,000 TPBARs	1.9×10^{-13}	2.1×10^{-15}	9.0×10^{-15}
	3,400 TPBARs	5.8×10^{-13}	6.4×10^{-15}	2.7×10^{-14}
Sequoyah	1,000 TPBARs	2.5×10^{-13}	2.0×10^{-15}	7.4×10^{-15}
	3,400 TPBARs	7.5×10^{-13}	6.1×10^{-15}	2.2×10^{-14}
Bellefonte	1,000 TPBARs	3.2×10^{-14}	1.7×10^{-15}	3.8×10^{-16}
	3,400 TPBARs	9.6×10^{-14}	5.1×10^{-15}	1.2×10^{-15}

[a] Increased likelihood of cancer fatality per year.

D.1.3.5 Rail Transportation Cask Handling Accident

The rail transportation cask handling accident source term and accident frequency data presented in Section D.1.1.7 were evaluated using the GENII accident analysis computer code (PNL 1988). Analyses were performed in accordance with guidance provided in NRC Regulatory Guide 4.2 (NRC 1976). **Table D–28** summarizes the consequences of the rail transportation cask handling accident to the maximally exposed offsite individual, an average individual in the public within an 80-kilometer (50-mile) radius of the reactor site, a noninvolved worker at the Watts Bar and Bellefonte Nuclear Plant sites located 640 meters (0.4 miles) from the release point, and a noninvolved worker at the Sequoyah Nuclear Plant located at the site boundary 556 meters (0.35 mile) from the release point. The risks associated with the rail transportation cask handling accident to the same receptors are summarized in **Table D–29**.

Table D–28 Rail Transportation Cask Handling Accident Consequences

Reactor Site	*Maximally Exposed Offsite Individual*		*Average Individual in Population to 80 kilometers (50 miles)*		*Noninvolved Worker*	
	Dose (rem)	*Cancer Fatality* [a]	*Dose (rem)*	*Cancer Fatality* [a]	*Dose (rem)*	*Cancer Fatality* [a]
Watts Bar	0.00072	3.6×10^{-7}	8.0×10^{-6}	4.0×10^{-9}	0.000045	1.7×10^{-8}
Sequoyah	0.00093	4.7×10^{-7}	7.5×10^{-6}	3.8×10^{-9}	0.000036	1.4×10^{-8}
Bellefonte	0.00012	6.0×10^{-8}	6.4×10^{-6}	3.2×10^{-9}	1.8×10^{-6}	7.2×10^{-10}

[a] Increased likelihood of cancer fatality.

Table D–29 Rail Transportation Cask Handling Accident Annual Risks

Reactor Site	*Tritium Production Core Configuration*	*Maximally Exposed Offsite Individual* [a]	*Average Individual in Population to 80 kilometers (50 miles)* [a]	*Noninvolved Worker* [a]
Watts Bar	1,000 TPBARs	9.7×10^{-14}	1.1×10^{-15}	4.6×10^{-15}
	3,400 TPBARs	2.9×10^{-13}	3.2×10^{-15}	1.4×10^{-14}
Sequoyah	1,000 TPBARs	1.3×10^{-13}	1.0×10^{-15}	3.8×10^{-15}
	3,400 TPBARs	3.8×10^{-13}	3.0×10^{-15}	1.1×10^{-14}
Bellefonte	1,000 TPBARs	1.6×10^{-14}	8.6×10^{-16}	1.9×10^{-16}
	3,400 TPBARs	4.8×10^{-14}	2.6×10^{-15}	5.8×10^{-16}

[a] Increased likelihood of cancer fatality per year.

D.1.3.6 Beyond Design-Basis Accident

The beyond design-basis accident source term and accident frequency data presented in Tables D–10, D–11, D–13, and D–14 were evaluated using the MACCS2 accident analysis computer code (SNL 1997). **Table D–30** summarizes the consequences of the beyond design-basis accident, with mean meteorological conditions, to the maximally exposed offsite individual and an average individual in the public within an 80-kilometer (50-mile) radius of the reactor site. The assessment of dose and the associated cancer risk to the noninvolved worker are not applicable for beyond design-basis accidents. A site emergency would have been declared early in the beyond design-basis accident sequence, and all nonessential site personnel would have evacuated the site in accordance with site emergency procedures before any radiological releases to the environment occurred. In addition, emergency action guidelines would be implemented to initiate evacuation of the public within 16.1 kilometers (10 miles) of the plant. The location of the maximally exposed offsite individual may or may not be at the site boundary for these accident sequences because emergency action guidelines would have been implemented and the population would be evacuating from the path of the radiological plume released by the accident. The MACCS2 computer code models the evacuation sequence to estimate the dose to the maximally exposed individual and the general population within 80 kilometers (50 miles) of the accident. The risks associated with the beyond design-basis accident to the same receptors are summarized in **Table D–31**.

Table D–30 Beyond Design-Basis Accident Consequences

Reactor Site	Tritium Production	Maximally Exposed Offsite Individual		Average Individual in Population to 80 kilometers (50 miles)		Noninvolved Worker	
		Dose (rem)	Cancer Fatality [a]	Dose (rem)	Cancer Fatality [a]	Dose (rem)	Cancer Fatality [a]
Release Category I - Vessel Breach with Early Containment Failure							
Watts Bar	0 TPBARs (No Action)	19.7	0.0099	0.25	0.00013	Not applicable	Not applicable
	1,000 TPBARs	19.7	0.0099	0.25	0.00013	Not applicable	Not applicable
	3,400 TPBARs	19.8	0.0099	0.25	0.00013	Not applicable	Not applicable
Sequoyah	0 TPBARs (No Action)	25.0	0.025	0.48	0.00024	Not applicable	Not applicable
	1,000 TPBARs	25.0	0.025	0.48	0.00024	Not applicable	Not applicable
	3,400 TPBARs	25.1	0.025	0.48	0.00024	Not applicable	Not applicable
Bellefonte	0 TPBARs [b]	2.3	0.0012	0.023	0.000012	Not applicable	Not applicable
	1,000 TPBARs	2.3	0.0012	0.023	0.000012	Not applicable	Not applicable
	3,400 TPBARs	2.4	0.0012	0.024	0.000012	Not applicable	Not applicable
Release Category II - Vessel Breach with Containment Bypass							
Watts Bar	0 TPBARs (No Action)	6.4	0.0032	0.35	0.00018	Not applicable	Not applicable
	1,000 TPBARs	6.4	0.0032	0.35	0.00018	Not applicable	Not applicable
	3,400 TPBARs	6.4	0.0032	0.35	0.00018	Not applicable	Not applicable
Sequoyah	0 TPBARs (No Action)	10.4	0.0052	0.72	0.00036	Not applicable	Not applicable
	1,000 TPBARs	10.4	0.0052	0.72	0.00036	Not applicable	Not applicable
	3,400 TPBARs	10.4	0.0052	0.73	0.00037	Not applicable	Not applicable
Bellefonte	0 TPBARs [b]	34	0.034	0.20	0.00010	Not applicable	Not applicable
	1,000 TPBARs	34	0.034	0.20	0.00010	Not applicable	Not applicable
	3,400 TPBARs	34	0.034	0.20	0.00010	Not applicable	Not applicable
Release Category III - Vessel Breach with Late Containment Failure							
Watts Bar	0 TPBARs (No Action)	0.51	0.00026	0.024	0.000012	Not applicable	Not applicable
	1,000 TPBARs	0.51	0.00026	0.025	0.000013	Not applicable	Not applicable
	3,400 TPBARs	0.53	0.00027	0.025	0.000013	Not applicable	Not applicable
Sequoyah	0 TPBARs (No Action)	0.84	0.00042	0.051	0.000026	Not applicable	Not applicable
	1,000 TPBARs	0.85	0.00042	0.052	0.000026	Not applicable	Not applicable
	3,400 TPBARs	0.87	0.00044	0.053	0.000027	Not applicable	Not applicable
Bellefonte	0 TPBARs [b]	0.37	0.00019	0.016	8.0×10^{-6}	Not applicable	Not applicable
	1,000 TPBARs	0.37	0.00019	0.016	8.0×10^{-6}	Not applicable	Not applicable
	3,400 TPBARs	0.38	0.00019	0.017	8.5×10^{-6}	Not applicable	Not applicable

[a] Increased likelihood of cancer fatality.

[b] The 0 TPBAR entry is included for consistency with the Watts Bar and Sequoyah Nuclear Plant analyses. The No Action Alternative at the Bellefonte Nuclear Plant implies that the reactors are not brought into commercial service. The No Action Alternative radiological dose is 0.

Table D–31 Beyond Design-Basis Accident Annual Risks

Reactor Site	*Tritium Production*	*Maximally Exposed Offsite Individual* [a]	*Average Individual in Population to 80 kilometers (50 miles)* [a]	*Noninvolved Worker*
Release Category I - Vessel Breach with Early Containment Failure				
Watts Bar	0 TPBARs (No Action)	6.7×10^{-9}	8.8×10^{-11}	Not applicable
	1,000 TPBARs	6.7×10^{-9}	8.8×10^{-11}	Not applicable
	3,400 TPBARs	6.7×10^{-9}	8.8×10^{-11}	Not applicable
Sequoyah	0 TPBARs (No Action)	1.7×10^{-8}	1.6×10^{-10}	Not applicable
	1,000 TPBARs	1.7×10^{-8}	1.6×10^{-10}	Not applicable
	3,400 TPBARs	1.7×10^{-8}	1.6×10^{-10}	Not applicable
Bellefonte	0 TPBARs [b]	1.1×10^{-9}	1.1×10^{-11}	Not applicable
	1,000 TPBARs	1.1×10^{-9}	1.1×10^{-11}	Not applicable
	3,400 TPBARs	1.1×10^{-9}	1.1×10^{-11}	Not applicable
Release Category II - Vessel Breach with Containment Bypass				
Watts Bar	0 TPBARs (No Action)	2.2×10^{-8}	1.2×10^{-9}	Not applicable
	1,000 TPBARs	2.2×10^{-8}	1.2×10^{-9}	Not applicable
	3,400 TPBARs	2.2×10^{-8}	1.2×10^{-9}	Not applicable
Sequoyah	0 TPBARs (No Action)	2.1×10^{-8}	1.4×10^{-9}	Not applicable
	1,000 TPBARs	2.1×10^{-8}	1.4×10^{-9}	Not applicable
	3,400 TPBARs	2.1×10^{-8}	1.5×10^{-9}	Not applicable
Bellefonte	0 TPBARs [b]	3.1×10^{-8}	9.1×10^{-11}	Not applicable
	1,000 TPBARs	3.1×10^{-8}	9.1×10^{-11}	Not applicable
	3,400 TPBARs	3.1×10^{-8}	9.1×10^{-11}	Not applicable
Release Category III - Vessel Breach with Late Containment Failure				
Watts Bar	0 TPBARs (No Action)	2.4×10^{-9}	1.1×10^{-10}	Not applicable
	1,000 TPBARs	2.4×10^{-9}	1.2×10^{-10}	Not applicable
	3,400 TPBARs	2.5×10^{-9}	1.2×10^{-10}	Not applicable
Sequoyah	0 TPBARs (No Action)	3.9×10^{-9}	2.4×10^{-10}	Not applicable
	1,000 TPBARs	3.9×10^{-9}	2.4×10^{-10}	Not applicable
	3,400 TPBARs	4.0×10^{-9}	2.5×10^{-10}	Not applicable
Bellefonte	0 TPBARs [b]	9.7×10^{-10}	4.1×10^{-11}	Not applicable
	1,000 TPBARs	9.7×10^{-10}	4.1×10^{-11}	Not applicable
	3,400 TPBARs	9.7×10^{-10}	4.3×10^{-11}	Not applicable

[a] Increased likelihood of cancer fatality per year.

[b] The 0 TPBAR entry is included for consistency with the Watts Bar and Sequoyah Nuclear Plant analyses. The No Action Alternative at the Bellefonte Nuclear Plant implies that the reactors are not brought into commercial service. The No Action Alternative radiological dose is 0.

D.2 HAZARDOUS CHEMICAL ACCIDENT IMPACTS ON HUMAN HEALTH

D.2.1 Accident Scenario Selection and Description

D.2.1.1 Accident Scenario Selection

Tritium production at the Watts Bar and Sequoyah Nuclear Plants would not introduce any additional operations that require the use of hazardous chemicals. No hazardous chemical accidents attributable to tritium production are postulated for the Watts Bar and Sequoyah Nuclear Plants.

The chemical inventory for Bellefonte was reviewed to identify potential accident scenarios. The chemical inventory at Bellefonte is given in **Table D–32** (TVA 1998):

Table D–32 Chemical Inventory at the Bellefonte Nuclear Plant Site

Location	*Chemical*	*Storage*	*Quantity per Tank (gallons)*
Auxiliary Building	Boric Acid	1 Tank	2,340
		1 Tank	18,700
		2 Tanks	31,400
	Sodium Hydroxide	2 Tanks[a]	16,500
	Hydrazine (35 percent)	1 Tank	100
	Lithium Hydroxide	1 Tank	70
	Sodium Hydroxide	1 Tank	210
	Sulfuric Acid	batteries	5,000
Turbine Building	Ammonium Hydroxide	1 Tank	140
		1 Tank	175
		1 Tank	300
		1 Tank	500
		1 Tank	525
		1 Tank	4,000
	Hydrazine (35 percent)	2 Tanks	110
		1 Tank	250
		1 Tank	300
		1 Tank	525
	Sodium Hydroxide	1 Tank	250
	Sulfuric Acid	1 Tank	250
Chemical Storage Building	Sodium Hydroxide	1 Tank	13,000
	Sulfuric Acid	1 Tank	13,000

[a] One tank for each unit.

The largest quantity of material at risk that is likely to volatilize and be dispersed following accidental release from the tanks is in the turbine building. The hazardous chemicals stored in the turbine building were reviewed against the Emergency Planning and Community Right-to-Know Act, Section 302, Extremely Hazardous Substances List Threshold Planning Quantity values published by the EPA (EPA 1996) to determine if the quantities of chemicals stored in the turbine building exceed the Threshold Planning Quantity threshold values. In the event that the inventory of a chemical exceeds the Threshold Planning Quantity value, the EPA requires that emergency response planning actions be conducted, including evaluation of potential accident scenarios. Only the chemical inventory in the Turbine Building was used for the purpose of this analysis. The physical properties of the other chemicals suggest that they would be of less concern with respect to widespread exposure upon accidental release from storage tanks. The inventory of two chemicals exceeded the Threshold Planning Quantity values. These Threshold Planning Quantity values are:

Ammonium Hydroxide Threshold Planning Quantity = 500 pounds for anhydrous ammonia
Hydrazine Threshold Planning Quantity = 1,000 pounds

D.2.1.2 Accident Scenario Descriptions

Two hazardous chemical accident scenarios are postulated for this EIS: (1) the accidental uncontrolled release of ammonium hydroxide, and (2) the accidental uncontrolled release of hydrazine.

Ammonium Hydroxide Release

EPA requires that the chemical accident analysis consider the release of the maximum inventory from the largest tank. The ammonium hydroxide release scenario was developed based on the following information:

- The largest ammonium hydroxide storage tank volume is 4,000 gallons (TVA 1998).
- The ammonium hydroxide storage tanks are located inside a room in the Turbine Building and are surrounded by an 828-square foot dike (TVA 1998).
- The ammonium hydroxide concentration is 30 percent ammonia by weight (TVA 1998).

The scenario assumes that a break occurs in the largest ammonium hydroxide storage tank, releasing the entire contents of the tank (4,000 gallons) inside the confined area in the room formed by the dike. The released material forms a pool with an effective area of 828 square feet. Ammonia then evaporates from the ammonium hydroxide liquid pool and forms a vapor cloud that fills the immediate area, leaks from the building, and moves downwind away from the building.

The rate of ammonia evaporation from a 30 percent concentration ammonium hydroxide pool is given in the *Draft Risk Management Program Guidance–Wastewater Treatment Facilities Hazard Assessment*, June 1998 (EPA 1998) as follows:

$$QR = 0.036A_p$$

where A_p is the diked area in square feet, and QR is the rate of evaporation in pounds per minute

Based on a pool area of 828 square feet, the rate of ammonia evaporation from the pool is:

$$QR = 0.036 \times 828 = 29.8 \text{ pounds per minute}$$

Hydrazine Release

The hydrazine release scenarios were developed for conditions similar to those described for the ammonium hydroxide release scenarios. However, the accident analysis computer code has the capability of modeling pool evaporation for pure chemicals such as hydrazine.

The scenario assumes the release of 525 gallons of hydrazine (35 percent concentration) inside the room of the Turbine Building. Although hydrazine is very reactive, the scenario does not assume any loss of the material by reactivity. The release is assumed to form a pool on the floor, with hydrazine vapor generated from pool evaporation. The vapor fills the immediate area, leaks from the building, and is dispersed downwind. The effective pool area is the same as that of the ammonium hydroxide release case (i.e., 828 square feet) because the tank is located within the same dike. Since hydrazine has a relatively high boiling point, no ground effect is assumed in the release scenario.

D.2.2 Chemical Accident Analysis Methodology

The potential health impacts from accidental releases of hazardous chemicals were assessed by comparing estimated airborne concentrations of the chemicals to Emergency Response Planning Guidelines developed by the American Industrial Hygiene Association. The Emergency Response Planning Guidelines values are not regulatory exposure guidelines and do not incorporate the safety factors normally included in healthy worker exposure guidelines. Emergency Response Planning Guideline-1 values are maximum airborne concentrations below which nearly all individuals could be exposed for up to one hour, resulting in only mild, transient, and reversible adverse health impacts. Emergency Response Planning Guideline-2 values are protective of irreversible or serious health effects or impairment of an individual's ability to take protective action. Emergency Response Planning Guideline-3 values are indicative of potentially life-threatening health effects.

Emergency Response Planning Guideline values have not been developed for ammonium hydroxide. Upon release of ammonium hydroxide from the storage tanks, ammonia will volatilize and be dispersed downwind to expose potential receptors. Therefore, the Emergency Response Planning Guideline values for ammonia were used to evaluate the potential health impacts of an ammonium hydroxide release. The Emergency Response Planning Guideline values for ammonia and hydrazine are presented in **Table D–33**.

Table D–33 Emergency Response Planning Guide Values for Hydrazine and Ammonia

Chemicals	*ERPG-1 (parts per million)*	*ERPG-2 (parts per million)*	*ERPG-3 (parts per million)*
Hydrazine [a]	0.03	8	80
Ammonia [b]	25	200	1000

ERPG = Emergency Response Planning Guide.
[a] Gephart, et al. 1994.
[b] Craig, et al. 1995.
Note: Hydrazine ERPGs were removed by the American Industrial Hygiene Association for further study in 1996 and have not been reinserted as of July 1998.

D.2.2.1 Receptor Description

The potential health impacts of the accidental release of ammonium hydroxide and hydrazine were assessed for two types of receptors:

- noninvolved workers - workers assumed to be located 640 meters from the point of release

- maximally exposed offsite individual - a member of the public located off site at the site boundary, 914 meters from the point of release

Facility workers (i.e. those individuals in the building at the time of the accident) were assumed to be killed by the release. The analysis took no credit for mitigative actions (e.g., area atmosphere monitoring, area evacuation alarms, emergency operating procedures) or accident precursors (e.g., leak before break) to reduce the accident consequences to the facility worker.

D.2.2.2 Analysis Computer Code Selection

The computer code selected for estimation of airborne concentrations is the Computer Aided Management of Emergency Operations (CAMEO)/Areal Locations of Hazardous Atmospheres (ALOHA), developed by the National Safety Council, the EPA, and the National Oceanic and Atmospheric Administration (NSC 1990).

D.2.2.3 Description of the Model

The atmospheric dispersion modeling for the above scenarios was conducted using the ALOHA 5.05 computer code (NSC 1990).

The ALOHA code was designed for use by first responders. The model is most useful for estimating plume extent and concentration downwind from the release source for short-duration chemical accidents. It uses a Gaussian dispersion model to describe the movement and spreading of a gas that is neutrally buoyant. For heavier-than-air vapor releases, the model uses the same calculations as those used in the DEGADIS model, an EPA heavy gas dispersion model (EPA 1989).

There are a number of limitations to the model, and these are summarized below:

- ALOHA is not intended for use with accidents involving radioactive chemicals.

- It is not intended for use with the permitting of stack gas or chronic, low-level (fugitive) emissions.

- The ALOHA-DEGADIS heavy gas module is more conservative than the DEGADIS model, which may result in a larger footprint than actually would be expected.

- ALOHA does not consider the effects of thermal energy from fire scenarios or the byproducts resulting from chemical reactions.

- ALOHA does not include the process needed to model particulate dispersion.

- ALOHA does not consider the shape of the ground under the spill or in the area affected by the plume.

- ALOHA does not estimate concentrations under very low wind speeds (less than 1 meter per second), since the wind direction may become inconsistent at these conditions.

- Under very stable atmospheric conditions (usually late night or early morning), the model estimates will have large uncertainties due to shifting wind directions and virtually no mixing of the plume into the surrounding air. Thus, these processes may lead to high airborne concentrations for long periods of time or at large distances from the release source.

- ALOHA does not accurately represent variations associated with near-field (close to the release source) patchiness. In the case of a neutrally buoyant gas, the plume will move downwind; but very near the source, the plume can be oriented in a different direction (such as going backward) due to the effect of drifting eddies in the wind.

D.2.2.4 Weather Condition Assumptions

The model results are presented for atmospheric Stability Classes D and F, with wind speeds of 5.3 meters per second and 1.5 meters per second, respectively. Atmospheric Stability Class D is considered to be representative of "average" weather conditions; Stability Class F is considered to be representative of "worst-case" weather conditions. These weather conditions were selected because they are recommended by the EPA in its *Technical Guidance for Hazards Analysis* (EPA 1987).

The model parameter values for these weather conditions are as follows:

1. Average Condition — Stability Class D
   Ambient air temperature: 75°F
   Relative humidity: 50 percent
   Cloud cover: 50 percent
   Average wind speed: 5.3 meters per second

2. Worst-Case Condition — Stability Class F
   Ambient air temperature: 60°F
   Relative humidity: 25 percent
   Cloud cover: 20 percent
   Average wind speed: 1.5 meters per second

D.2.3 Human Health Impacts

The potential health impacts from the accidental releases were assessed by comparing the modeled ambient concentrations of ammonia and hydrazine at each of the receptor locations identified previously to the Emergency Response Planning Guidelines. The estimated airborne concentrations of ammonia and hydrazine are presented in **Table D–34** and **Table D–35** respectively. **Table D–36** presents a summary of the impacts data.

D.2.3.1 Impacts to Noninvolved Workers

Noninvolved workers are assumed to be located at 640 meters from the point of release. The concentrations of ammonia at 640 meters range from 14 to 318 parts per million, based on the assumed meteorological conditions. The maximum estimated airborne concentration at 640 meters in the F stability class exceeds the Emergency Response Planning Guideline-2 value of 200 parts per million for ammonia, which suggests that noninvolved workers may experience irreversible or serious, but not life-threatening, adverse health effects if the exposures are not mitigated.

For the hydrazine release scenarios, the concentrations at 640 meters range from 0.8 to 6.0 parts per million, based on the assumed meteorological conditions. As a result, the maximum estimated airborne concentration at 640 meters exceeds the Emergency Response Planning Guideline-1 value of 0.03 parts per million for hydrazine, which suggests the potential for only mild, transient, and reversible adverse health impacts to noninvolved workers.

Table D–34 Airborne Concentration Estimates for Ammonium Hydroxide (NH_3)Release Scenarios

Downwind Distance from Source (meters)	*NH_3 Concentration under Stability Class D*		*NH_3 Concentration under Stability Class F*	
	milligrams per cubic meters	*(parts per million)*	*milligrams per cubic meters*	*(parts per million)*
30	3,233	(4,590)	83,900	(119,138)
100	306	(435)	7,730	(10,976)
500	15.5	(22)	352	(500)
640	9.9	(14)	224	(318)
914	5.4	(7.7)	119	(169)
1000	4.7	(6.7)	102	(145)
1500	2.5	(3.5)	51.6	(73)
2000	1.5	(2.2)	32.7	(46)

Table D–35 Airborne Concentration Estimates for Hydrazine Release Scenarios

Downwind Distance from Source(meters)	*Concentration under Stability Class D*		*Concentration under Stability Class F*	
	milligrams per cubic meters	*(parts per million)*	*milligrams per cubic meters*	*(parts per million)*
30	168	(127)	730	(561)
100	30	(22.7)	194	(149)
500	1.6	(1.2)	12.2	(9.4)
640	1.1	(0.8)	7.81	(6.0)
914	0.5	(0.4)	4.17	(3.2)
1000	0.5	(0.4)	3.56	(2.7)
1500	0.3	(0.2)	1.7	(1.3)
2000	--	--	1.07	(0.8)

Table D–36 Summary of Impacts Data for Release Scenarios

	Guidelines	*Hydrazine (Stability Class D)*	*Hydrazine (Stability Class F)*	*Ammonia (Stability Class D)*	*Ammonia (Stability Class F)*
	ERPG-1 ERPG-2 ERPG-3	>2000 179 44	>2000 500 200	464 150 65	2250 825 425
Noninvolved worker (640 meters)	Parts per million Level of concern Potential health effects	0.8 ERPG-1 Mild, transient	6 ERPG-1 Mild, transient	16 ERPG-1 Mild, transient	318 ERPG-2 Serious
Maximally exposed offsite individual (914 meters)	Parts per million Level of concern Potential health effects	0.4 ERPG-1 Mild, transient	3.2 ERPG-1 Mild, transient	7.7 ERPG-1 None (<ERPG-1)	169 ERPG-1 Mild, transient

ERPG = Emergency Response Planning Guideline.

D.2.3.2 Offsite Impacts

The maximally exposed offsite individual is assumed to be located at a distance of 914 meters from the point of release. For the ammonium hydroxide release scenarios, the offsite receptor will be potentially exposed to an ammonia concentration of 7.7 parts per million under Stability Class D condition (see Table D–34), which is below the Emergency Response Planning Guideline-1 value for ammonia of 25 parts per million. Exposures to concentrations below the Emergency Response Planning Guideline-1 value are not expected to produce any adverse health effects for the offsite receptor. Under Stability Class F conditions, the offsite receptor may be exposed to an ammonia concentration of about 169 parts per million which is below the Emergency Response Planning Guideline-2 value for ammonia of 200 parts per million. Exposure of the offsite receptor at concentrations greater than the Emergency Response Planning Guideline-1 value but less than the Emergency Response Planning Guideline-2 value may produce only mild, transient and reversible adverse health effects.

For the hydrazine release scenarios, the offsite receptor exposure concentrations range from 0.4 parts per million to 3.2 parts per million (see Table D–35; both stability classes). These concentrations exceed the Emergency Response Planning Guideline-1 value for hydrazine of 0.03 parts per million, but are less than the Emergency Response Planning Guideline-2 value of 8 parts per million. This suggests that the offsite receptor may experience only mild, transient, and reversible adverse health effects as a result of the exposure.

D.2.3.3 Uncertainties in the Dispersion Analyses

The results of this screening level analysis contain a number of uncertainties in the atmospheric dispersion calculations, some of which are summarized below:

- The dispersion modeling does not take into account the reduction in the predicted rate of evaporation because the spillage is inside the building; the dilution is caused by the structures on the site; or the potential for other mitigating actions. There are no accurate methods for predicting the extent of this dilution, but predicted concentrations at any point could well be too high by factors of 2 to 5 or more.

- The dispersion modeling does not take account of the deposition of highly reactive vapors (such as hydrazine) onto surfaces including equipment, the ground, water, and vegetation. This means that the model overestimates airborne concentrations at longer distances.

- Overall, the uncertainties in predicted airborne concentrations may be as large as a factor of $\pm 2 \times$ the estimated concentration.

In view of these uncertainties, the results of this analyses should be considered only as screening level estimations. TVA will conduct analyses to comply with requirements specified in 40 CFR 68 prior to operation of the Bellefonte Nuclear Power Plant.

D.3 REFERENCES

AEC (U.S. Atomic Energy Commission), 1972, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure Safety Guide 24*, (NRC/RGS 1.24), Washington, DC, March 23.

Craig, D. K., J. S. Davis, R. DeVore, D. J. Hanson, A. J. Petrocehi, T. J. Powell, 1995, "Alternative Guideline Limits for Chemicals without Environmental Response Planning Guidelines," *American Industrial Hygiene Association Journal,* 56: 919–925.

CRC (CRC Press), 1982, *CRC Handbook of Radiation Measurement and Protection*, Section A, Volume II: Biological and Mathematical Information, CRC Press.

DOC (U.S. Department of Commerce), 1992, *1990 Census of Population and Housing, Summary Tape File 3 on CD ROM Technical Documentation*, Bureau of Census, Washington, DC, May.

DOC (U.S. Department of Commerce), 1993, *1992 Census of Agriculture for Selected Counties in Tennessee, Alabama, Georgia, and North Carolina*.

DOE (U.S. Department of Energy), 1996, *Commercial Light Water Reactor Systems Description and Requirements Document (SDRD)*, CLWR-RD-96-5681(08)-06, Revision 0, Defense Programs Office, Washington, DC, October.

Duke (Duke Power Company), 1990, *Oconee Nuclear Station Units 1, 2, 3 Individual Plant Examination Submittal Report*, Charlotte, North Carolina, December.

Entergy (Entergy Operation Inc.), 1993, *Arkansas Nuclear One, Unit 1 Probabilistic Risk Assessment - Individual Plant Examination Submittal for Arkansas Nuclear One, Unit 1*, Docket No. 50-0313, April.

EPA (U.S. Environmental Protection Agency), 1987, *Technical Guidance for Hazards Analysis*, Federal Emergency Management Agency and U.S. Department of Transportation, Washington, DC, December.

EPA (U.S. Environmental Protection Agency), 1988, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, Federal Guidance Report 11, EPA 520/1-88-020, Washington DC, September.

EPA (U.S. Environmental Protection Agency), 1989, *User's Guide for the DEGADIS 2.1 Dense Gas Dispersion Model*, EPA-450/4-89-019, U.S. Environmental Protection Agency, Office of Air Quality Planning and Standards, Office of Air and Radiation, Research Triangle Park, North Carolina, November.

EPA (U.S. Environmental Protection Agency), 1993, *External Exposure to Radionuclides in Air, Water, and Soil*, Federal Guidance Report 12, EPA 402-R-93-081, Washington DC.

EPA (U.S. Environmental Protection Agency), 1996, Title III List of Lists, EPA-550-B-96-015, December.

EPA (U.S. Environmental Protection Agency), 1998, *Draft Risk Management Program Guidance–Wastewater Treatment Facilities Hazard Assessment,* June.

Gephart, L., D. P. Kelly, F. L. Cavender, G. M. Rusch, 1994, "Emergency Response Planning Guidelines," *American Industrial Hygiene Association Journal,* 55 (June): 560–561.

GPUN (GPU Nuclear) 1993, *TMI Unit 1 Individual Plant Examination Submittal Report*, Docket No. 50-0289, March.

ICRP (International Commission on Radiological Protection), 1980, *Limits for Intakes of Radionuclides by Workers, Annals of the ICRP,* ICRP Publication 30, Pergamon Press, New York, New York.

ICRP (International Commission on Radiological Protection), 1983, *Radionuclide Transformations: Energy and Intensity of Emissions, Annals of the ICRP*, ICRP Publication 38, Pergamon Press, New York, New York.

ICRP (International Commission on Radiological Protection), 1986, *The Metabolism of Plutonium and Related Elements, Annals of the ICRP,* ICRP Publication 48, Pergamon Press, New York, New York.

NCRP (National Council on Radiation Protection and Measurements), 1993, *Risk Estimates for Radiation Protection*, NCRP Report No. 115, Bethesda, Maryland, December 31.

NRC (U.S. Nuclear Regulatory Commission), 1974, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant for Pressurized Water Reactors*, Regulatory Guide 1.4, Revision 2, June.

NRC (U.S. Nuclear Regulatory Commission), 1975, *Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, WASH-1400 (NUREG-75/014), Washington, DC, October.

NRC (U.S. Nuclear Regulatory Commission), 1976, *Preparation of Environmental Reports for Nuclear Power Stations*, NUREG-0099, Regulatory Guide 4.2, Revision 2, Office of Standards Development, Washington, DC, July.

NRC (U.S. Nuclear Regulatory Commission),1977, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix, I, Regulatory Guide 1.109,* Revision 1, October.

NRC (U.S. Nuclear Regulatory Commission), 1983, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Regulatory Guide 1.145, Revision 1,* November.

NRC (U.S. Nuclear Regulatory Commission), 1987, *Severe Accidents in Spent Fuel Pool in Support of Generic Safety Issue 82,* NUREG/CR-4982, Washington, DC, July.

NRC (U.S. Nuclear Regulatory Commission) 1988, *Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f) (Generic Letter No. 88-20)*, Washington, DC, November 23.

NRC (U.S. Nuclear Regulatory Commission), 1990a, *MELCOR Accident Consequence Code System (MACCS)*, NUREG/CR-4691, February.

NRC (U.S. Nuclear Regulatory Commission), 1990b, *Evaluation of Severe Accident Risks: Sequoyah, Unit 1*, NUREG/CR-4551, Volume 5, Revision 1, Division of Systems Research, December.

NRC (U.S. Nuclear Regulatory Commission), 1991, *Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR §50.54(f) (Generic Letter No. 88-20, Supplement 4),* Washington, DC, June 28.

NSC (National Safety Council), 1990, *Areal Locations of Hazardous Atmospheres (ALOHA) Software*, Version 5.05, National Safety Council, Washington, DC, December.

PNL (Pacific Northwest Laboratory), 1988, *GENII--The Hanford Environmental Radiation Dosimetry Software System*, PNL-6584, Richland, Washington, November.

PNNL (Pacific Northwest National Laboratory), 1997, letter from Walter W. Laity to Stephen M. Sohinki, U.S. Department of Energy, "CLWR - Tritium Permeation," Richland, Washington, October 8.

PNNL (Pacific Northwest National Laboratory), 1999, letter from Walter W. Laity to Stephen M. Sohinki, U.S. Department of Energy, "TPBAR Tritium Releases Assumptions for the TPBAR EIS," February 10.

SNL (Sandia National Laboratory), 1997, *Code Manual for MACCS2: Volume 1, User's Guide*, SAND97-0594, Albuquerque, New Mexico, March.

TVA (Tennessee Valley Authority), 1991, *Bellefonte Nuclear Plant Final Safety Analysis Report*, through Amendment 30, Chattanooga, Tennessee, December 20.

TVA (Tennessee Valley Authority), 1992a, *Sequoyah Nuclear Plant Unit 1 Probabilistic Risk Assessment Individual Plant Examination Submittal*, Revision 0, Docket Numbers 50- 0327, and 50-0328, September.

TVA (Tennessee Valley Authority), 1992b, *Watts Bar Nuclear Plant Individual Plant Examination Submittal*, Revision 0, Docket No. 50-0390, September.

TVA (Tennessee Valley Authority), 1994, *Watts Bar Unit 1 Individual Plant Examination Update, Delta Report for Revision 1 Update*, Revision 1, Docket No. 50-0390, April.

TVA (Tennessee Valley Authority), 1995a, *Watts Bar Nuclear Plant Final Safety Analysis Report*, through Amendment 91, Chattanooga, Tennessee, October 24.

TVA (Tennessee Valley Authority), 1995b, *Sequoyah Nuclear Plant Individual Plant Examination, Delta Report for Revision 1 Update*, September.

TVA (Tennessee Valley Authority), 1996, *Sequoyah Nuclear Plant Updated Final Safety Analysis Report*, through Amendment 12, Chattanooga, Tennessee, December 6.

TVA (Tennessee Valley Authority), 1998, data collected from TVA personnel by Science Applications International Corporation personnel, January–August.

WEC (Westinghouse Electric Corporation), 1998a, letter from M. L. Travis to C. D. Harmon, Sandia National Laboratory, "Data Requested by SAIC for the EIS of Tritium Production in CLWRs," NDP-MLT-98-121, Pittsburgh, Pennsylvania, January 23.

WEC (Westinghouse Electric Corporation), 1998b, "Tritium Production Core (TPC) Topical Report" Unclassified, Non-proprietary version), NDP-98-181, July.

WHC (Westinghouse Hanford Company), 1991, *Light Water Reactor (WNP-1) Plant Description - New Production Reactor*, WHC-EP-0263, Revision 2, Richland, Washington, March.

WHC (Westinghouse Hanford Company), 1992, *WNP-1 New Production Reactor Limited-Scope Probabilistic Risk Assessment*, WHC-SP-0636, Richland Washington, January.